Exelon Nuclear
200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

Exelon Nuclear

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April 23, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Peach Bottom Atomic Power Station, Units 2 & 3
Facility Operating License Nos. DPR-44 and DPR-56
NRC Docket Nos. 50-277 and 50-278

Subject: Supplement to the Request for License Amendments Related to Application of Alternative Source Term, dated July 14, 2003

References:
(1) Letter from M. P. Gallagher (Exelon Generation Company, LLC) to US NRC, dated July 14, 2003
(2) Letter from G. F. Wunder (U. S. Nuclear Regulatory Commission) to J. L. Skolds (Exelon Generation Company, LLC), dated January 16, 2004
(3) Letter from M. P. Gallagher (Exelon Generation Company, LLC) to US NRC, dated March 15, 2004

This letter is being sent to supplement the License Amendment Request (LAR) to support application of an alternative source term (AST) methodology (Reference 1) at Peach Bottom Atomic Power Station (PBAPS), Units 2 & 3. This LAR proposed certain TS and TS Bases changes for PBAPS Units 2 & 3 as part of implementing an AST methodology.

In the Reference (2) letter, the U. S. Nuclear Regulatory Commission requested additional information. In the Reference (3) letter, Exelon provided a partial response to the request for additional information. Attachment 1 to this supplemental letter provides the response to the remaining questions associated with the request for additional information. Attachment 2 to this supplemental letter provides the revised TS Page Markups and TS Markup Inserts pages. Attachment 3 to this supplemental letter provides the revised TS Bases inserts. Attachment 4 to this supplemental letter provides the revised Camera-ready TS pages. Attachment 5 to this supplement provides the revised Camera-ready TS Bases pages. Attachment 6 provides the Regulatory Guide 1.183 Comparison Table Revision. Attachment 7 provides the Post-Accident Vital Area Access Considerations Table Revision from the original submittal. Attachment 8 provides the LOCA Radiological Consequences Analysis Revision from the original submittal.

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There is no impact to the No Significant Hazards Consideration submitted in the Reference 1 letter. There are no additional commitments contained within this letter.

If you have any questions or require additional information, please contact Doug Walker at (610) 765-5726.

I declare under penalty of perjury that the foregoing is true and correct.

Respectfully,

Executed on 04-23-04

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
Exelon Generation Company, LLC

Attachments:
1. Exelon Response to the Request for Additional Information
2. Revised TS Pages Markups and TS Markup Inserts pages
3. Revised TS Bases Inserts
4. Revised Camera-ready TS pages
5. Revised Camera-ready TS Bases pages
6. Regulatory Guide 1.183 Comparison Table Revision
7. Post-Accident Vital Area Access Considerations Table Revision
8. LOCA Radiological Consequences Analysis Revision

cc: H. J. Miller, Administrator, Region I, USNRC
C. W. Smith, USNRC Senior Resident Inspector, PBAPS
G. F. Wunder, Senior Project Manager, USNRC (by FedEx)
R. R. Janati - Commonwealth of Pennsylvania

ATTACHMENT 1

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Response to Request for Additional Information

REQUEST FOR ADDITIONAL INFORMATION
PEACH BOTTOM ATOMIC POWER STATION, UNITS 2 AND 3
PROPOSED USE OF ALTERNATIVE SOURCE TERM (AST) METHODOLOGY

The following questions remain from the January 16, 2004 NRC letter regarding the Peach Bottom Alternative Source Term RAI: 8, 11, 12, 13, 14, 19, 20, 21, 22, 23, 26, 27, and 28

Question #

8. Both Regulatory Guides (RG) 1.145 (Section 5.3) and 1.194 (Section 2) imply that the period with the most adverse release of radioactive materials to the environment should be assumed to occur coincident with the period of most unfavorable atmospheric dispersion. For the main stack releases, the highest control room X/Q values are associated with 0-2 hour flow reversal conditions and the highest offsite X/Q values are associated with the 0-0.5 hour fumigation conditions. Please describe how these highest X/Q values were used coincident with the most limiting portion of the release to the environment to estimate control room and offsite doses.

RESPONSE:

The 0–2 hour Offgas Stack χ/Q value of 2.72E-06 $^{sec}/_{m^3}$ was assigned to represent the Control Room distance of 209 meters. This value was predicted by the NRC model PAVAN, and was calculated actually for a distance of 500 meters. In accordance with Regulatory Guide 1.194 (formerly DG-1111), Section 3.2.2, the PAVAN model was executed, in addition to running ARCON96 since the Control Room is relatively close to the base of the tall Offgas Stack, and the ARCON96 model had produced negligibly small χ/Q values at the Control Room intake distance. Pursuant to Regulatory Guide 1.194, several distances in addition to the actual 209 m distance were modeled by PAVAN until the maximum 0-2 hour χ/Q was determined, as predicted to occur at 500 meters. This maximum χ/Q was then assigned as the 0-2 hour Control Room χ/Q value. As indicated by Regulatory Guide 1.194, this conservative procedure was performed to account for the possible diurnal wind direction changes, meander, or stagnation.

In the current submittal, the worst two hours were identified only for the EAB. The 0.5-hr fumigation condition was applied to the beginning of the 2-hour period, rather than the end. This has been corrected in new results submitted.

For LPZ and the Control Room (CR), the cumulative dose is typically dominated by later periods, in which case applying the worst χ/Q s to short maximum release periods would not be necessary.

We also note that the RADTRAD Version 3.03 deleted the Control Room and LPZ "worst 2 hour" period identification in the output. Nonetheless, since the LOCA analysis was modified, maximum (0.5 hr and 2 hr) χ/Qs has been applied to the maximum release periods for the EAB, LPZ, and CR unless otherwise justified.

11. The proposed revised UFSAR text identifies a change in methodology regarding how the containment leakage is addressed in the MCPA analysis.

    a. Provide the MCPA and containment overpressure license (COPL) calculation for the NRC staff's review.

    b. How is it different from the previously reviewed method described in PECO Energy Company's Calculation PM-1013, "Minimum Containment Pressure Calculation," Revision 3, February 2000?

    c. How are the main steam isolation valve (MSIV) and airlock leakages included in the calculation?

    d. How are the leakages conservatively varied with the containment pressure assuming turbulent flow?

RESPONSE:

(a) Exelon is providing the following information from the calculation to address the issues identified above. This represents the relevant information presented in the calculation relative to containment leakage. Arrangements can be made for the review of Exelon Nuclear Design Analysis PM-1013, "Minimum Containment Pressure Available", Revision 5, if required.

(b)(c)(d) The previous containment leakage assumptions considered the Technical Specification 0.5% per day leakage as remaining constant over time, throughout the entire event. The constant nitrogen leak rate from the containment was (Equation 2 from PM-1013, Revision 3):

$$\dot{m} = \frac{144 * (P_l + P_{atm}) * L_a * V}{24 * 3600 * R_a * (T_l + T_0)}$$

where :

$\dot{m}$ = nitrogen leakage mass flow, lbm/sec

P_l = nitrogen leakage pressure, psig

P_{atm} = atmospheric pressure, psia

L_a = nitrogen volumetric leakage rate, % per day

V = containment free volume, ft3

R_a = nitrogen ideal gas constant, (ft-lbf)/(lbm-°R)

T_l = nitrogen leakage temperature, °F

T_0 = temperature conversion, °F to °R

Values were selected to maximize the leakage (i.e., maximum P_l and minimum T_l). Only nitrogen is assumed to be expelled during the event.

The revised containment leakage methodology is based on the proposed PBAPS Technical Specification limit of 0.7% weight per day for general containment leakage at the test pressure of 49.1 psig, plus the proposed PBAPS Technical Specification limit of 158 scfh or less[1] for total MSIV leakage at a test pressure of 25 psig, plus the current PBAPS Technical Specification limit of 9000 scc per minute for airlock seal leakage at a test pressure of 49.1 psig. MSIV and airlock leakages are converted to the equivalent percent weight per day as follows (Assumption 5.J.x from PM-1013, Revision 5):

$$L_{MSIV} = \frac{24 \cdot 144 \cdot Q_{MSIV} \cdot P_{atm}}{R_N \cdot (60 + T_0) \cdot M_{a_i}} \cdot \sqrt{\frac{\Delta P_{ref}}{\Delta P_{test}}}$$

$$L_{airlock} = \frac{24 \cdot 60 \cdot 144 \cdot Q_{airlock} \cdot P_{atm}}{30.48^3 \cdot R_N \cdot (60 + T_0) \cdot M_{a_i}} \cdot \sqrt{\frac{\Delta P_{ref}}{\Delta P_{test}}}$$

where :

L_{MSIV} = MSIV leak rate, % weight per day at ΔP_{ref}

$L_{airlock}$ = airlock leak rate, % weight per day at ΔP_{ref}

Q_{MSIV} = MSIV leak rate, scfh

$Q_{airlock}$ = airlock leak rate, sccm

P_{atm} = atmospheric pressure, psia

R_N = gas constant for Nitrogen

M_{a_i} = initial containment Nitrogen mass, lbm

ΔP_{ref} = reference containment differential pressure, 49.1 psid

ΔP_{test} = test reference differential pressure, psid

T_0 = temperature conversion, °F to °R

Using the PBAPS Technical Specification leakage limits and the above expressions, L_{MSIV} and $L_{airlock}$ are calculated as 2.31% and 0.20% per day, respectively. Combining with the 0.7% per day general containment leakage, a total containment leak rate of 3.21% per day is estimated. Use of the assumed total containment leak rate of 3.21% per day, i.e., with each component at its maximum value, assumed to occur at the same time, is conservative.

With these containment leakage values normalized to the reference containment differential pressure, the % weight leakage at any given time is assumed to be a function of containment pressure:

$$L(t) = M_{ai} \cdot \frac{(L_{containment} + L_{MSIV} + L_{airlock})}{24 \cdot 3600} \cdot \Delta t \cdot \sqrt{\frac{\Delta P_d(t)}{\Delta P_{ref}}}$$

[1] The License Amendment Request proposed an MSIV leakage of 174 scfh. Currently, the MSIV leakage rate assumed for AST purposes is under discussion with NRC staff. Therefore, a maximum value of 158 scfh has been assumed for MCPA purposes, and will be confirmed subsequent to approval of the amendment request.

$$
\begin{aligned}
&where: \\
L(t) &= \text{total leakage during } \Delta t, \text{ lbm} \\
M_{ai} &= \text{initial containment Nitrogen mass, lbm} \\
L_{containment} &= \text{containment leak rate, \% weight per day at } \Delta P_{ref} \\
L_{MSIV} &= \text{MSIV leak rate, \% weight per day at } \Delta P_{ref} \\
L_{airlock} &= \text{airlock leak rate, \% weight per day at } \Delta P_{ref} \\
\Delta t &= \text{time step size, seconds} \\
\Delta P_d(t) &= \text{containment differential pressure, psid} \\
\Delta P_{ref} &= \text{reference differential pressure, 49.1 psid}
\end{aligned}
$$

The turbulent (orifice) flow relationship of flow being proportional to the square root of the pressure difference was used based on a review of TID-20583, Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations. Per TID-20583, to extrapolate downward from a high test pressure to a low actual pressure, the assumption of orifice flow should be used since it will result in the least change in leakage rate.

12. Previously, containment leakage was assumed to be constant at L_a=0.5%/day throughout the event. The containment leakage has been increased to L_a=0.7%/day for the first 24 hours, based on the proposed change to TS 5.5.12, for a peak post-accident containment pressure of 49.1 psig. This leakage is then reduced to 0.56xL_a=0.392%/day from 24 to 38 hours and then reduced to 0.50xL_a=0.350%/day, for 38 to 720 hours. In addition, MSIV leakage of 174 scfh is included (based on the proposed change to TS 3.6.1.3) in the MCPA calculation, with leakage measured at a test pressure of 25 psig. After 24 hours, the MSIV leak rate is reduced to 77.2%, then to 65.4% at 48 hours, to 59.0% at 72 hours, to 55.5% at 96 hours, and finally to 50% at 157 hours for the remainder of the event. Leakage from the personal airlock of 9,000 sccm, for a peak post-accident containment pressure of 49.1 psig, is also included in the proposed change to the MCPA calculation.

    a. How are the leakages conservatively varied with the containment pressure assuming turbulent flow?

    b. How does this evaluation differ from the MCPA and COPL calculation in question 11 above, which is only carried out to 12.5 hours?

    c. Identify the TS which controls the allowable airlock leakage rate.

RESPONSE:

The leakage values stated in the question above are only used to support dose analysis and are governed by the guidance relative to containment leakage in Regulatory Guide 1.183. These leakage assumptions are not the same as those used in the determination of the MCPA in calculation PM-1013, which does not support, take input from nor provide input to the dose calculations.

(a) As described in the response to NRC RAI Question #11 above, the reference containment leakage is assumed to be 3.21% weight per day at a reference containment pressure of 49.1 psig. In addition to the 0.7% weight per day general containment leakage, this also includes the leakage from the MSIVs and personnel airlock (L_{MSIV} and $L_{airlock}$ are 2.31%, 0.20% weight per day, respectively at ΔP_{ref}). During the analysis of the MCPA, the % weight leakage is varied only as a function of the containment pressure:

$$L(t) = M_{ai} \cdot \frac{(L_{containment} + L_{MSIV} + L_{airlock})}{24 \cdot 3600} \cdot \Delta t \cdot \sqrt{\frac{\Delta P_d(t)}{\Delta P_{ref}}}$$

(b) The MCPA analysis applies the leakage methodology described above out to 10,000,000 seconds (2777.8 hours) which corresponds to the duration of the GE analysis for the DBA LOCA long term suppression pool temperature response. Calculation PM-1013 provides documentation of this analysis through 13.8 hours to ensure the point of peak suppression pool temperature and peak MCPA is adequately covered.

(c) Peach Bottom Technical Specification 5.5.12 requires that overall airlock leakage is ≤ 9000 scc/min when tested at ≥ P_a. This requirement is verified by Surveillance Requirement 3.6.1.2.1.

13. During the previous amendment review (Hutton, J. A., PECO Energy Company, to NRC, "Peach Bottom Atomic Power Station, Units 2 and 3 Response to May 10, 2000, Telephone Questions Regarding PECO Energy License Amendment Request Related to Generic Letter 97-04," June 29, 2000) it was stated that the margin between the MCPA and COPL was set at 1 foot (0.42 psid). The proposed amendment would decrease this margin to about 0.28 psid.

    a. Provide a justification for reducing this agreed to margin.

    b. Provide a comparison of the COPL value to the COPR (containment overpressure required) value for the residual heat removal (RHR) and core spray pumps for the most limiting event(s), including the margin to the COPL value before and after the proposed change to the MCPA/COPL calculation.

While not directly related to the MCPA calculation, justification for the inclusion of the suppression chamber air space in the mixing of the radioactive release needs to be provided.

RESPONSE:

The COPL was established in reviews culminating in the previously mentioned NRC SER (Letter from B.C. Buckley, Sr., USNRC to J.A. Hutton, PECO Energy Company, August 14, 2000, "Peach Bottom Atomic Power Station, Unit Nos. 2 and 3 - Issuance of Amendment Regarding Crediting of Containment Overpressure for Net Positive Suction Head Calculations For Emergency Core Cooling Pumps (TAC Nos. MA6291 and MA6292)"). From that SER:

> *"The NRC staff performed confirmatory calculations of the RHR NPSH analysis. According to our calculations, the minimum margin between the COPL and the COPR for the RHR pumps is 0.88 psig. This occurred at the peak suppression pool temperature of 205.7°F. This margin allows for minor design changes which could affect the COPR. This result is consistent with the licensee's calculations. Additionally, our calculations demonstrated that the minimum margin between the COPL and MCPR was approximately 0.42 psig (1 foot). Because of the way the COPL was defined, i.e., the COPL will be 1 foot less than the MCPA for a design basis LOCA, this minimum margin is maintained over the entire COPL curve."*

and:

> *"For the long term following a LOCA, the staff has approved the use of the containment overpressure depicted on UFSAR Figure 5.2.16 and provided in the table above for both the RHR and core spray pumps."*

Since that time, some plant changes have been made which were not considered within the original intent of a minor design change. These changes included increases in the TS allowable river water temperature from 90°F to 92°F, correction of decay heat errors identified by GE in SIL-636 rev.1, the formal incorporation of 2σ decay heat uncertainty in the containment calculations, and the currently proposed increases in MSIV and containment leakage as part of AST. These changes had the net effect of decreasing the MCPA. The change in MCPA methodology necessary to accommodate AST proposed leakages, and its potential impact to COPL, is the very reason Exelon Nuclear has requested this NRC review.

(a) Although the COPL line was derived using a 1 foot margin to the MCPA, it is our understanding that, like the original Peach Bottom FSAR containment overpressure limit line, the NRC SER has established the COPL line itself, "depicted on UFSAR Figure 5.2.16" as the limit, rather than the maintaining of a specified margin to the MCPA. With the COPL being a fixed line, the proposed AST changes would have reduced the MCPA margin to the previous COPL from 1 foot of head (7.41 - 6.99=0.42 psid) to essentially zero (7.04 - 6.99=0.05 psid). Consequently, a new COPL limit needs to be proposed.

If instead of COPL being a fixed line, maintaining a 1 foot of head margin to the MCPA were the case, adequate overpressure margin would still be available to satisfy the NPSH requirements of the RHR pump during the design basis LOCA. A 1 foot of head margin to the new MCPA would produce a peak COPL of 6.62psig (7.04 - 0.42=6.62 psig), which still provides another foot of margin to the peak COPR for RHR of 6.14 psig (6.62 - 6.14=0.48 psid).

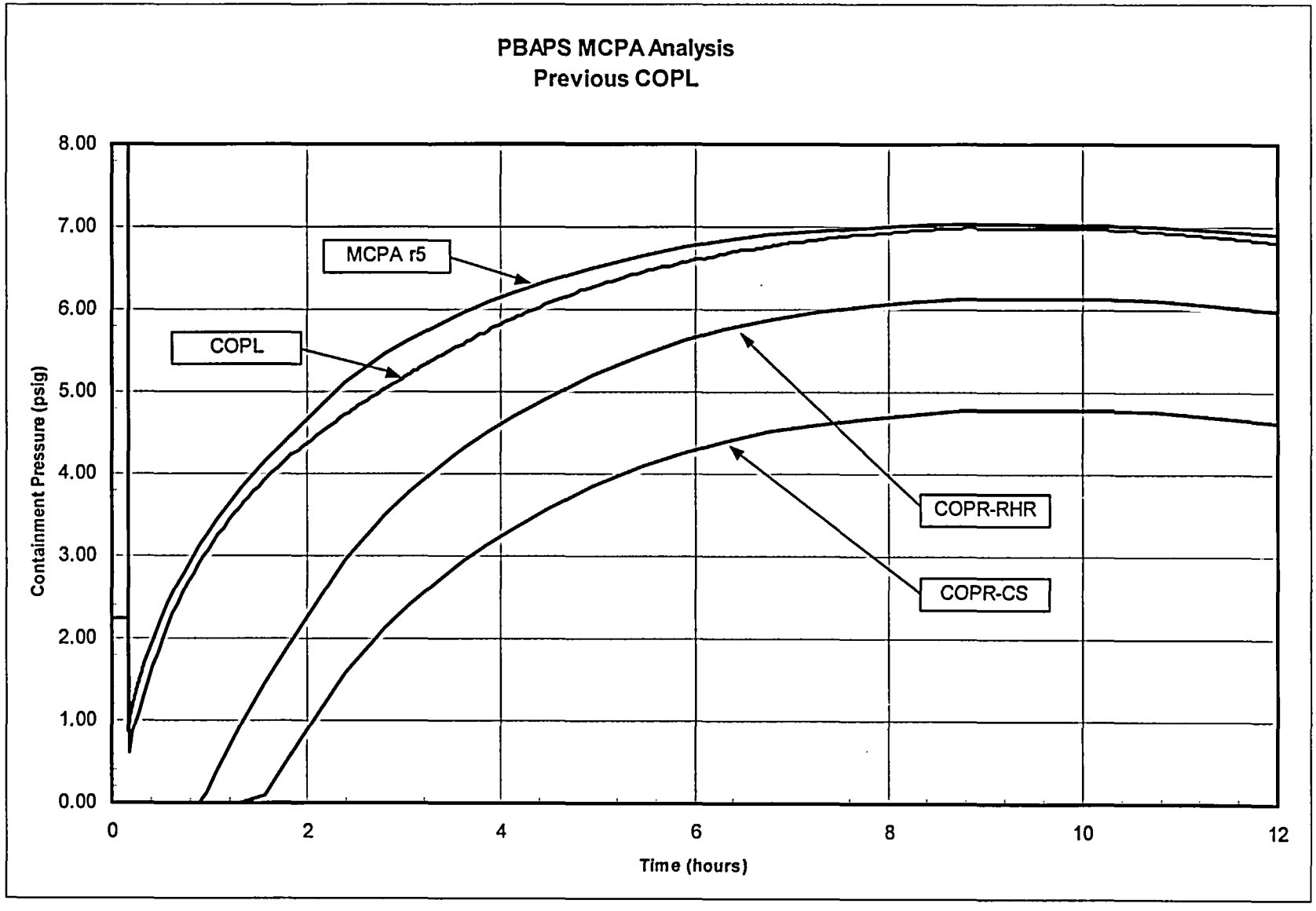
The COPL line that was proposed with the AST amendment request preserved the relative relationship between the COPR, COPL, and MCPA from the previous NRC review. With about a third of the available margin between MCPA and COPR being assigned to the COPL, the new proposed COPL line still maintains about a third of the available margin to the MCPA (0.69 foot, approx. 0.29 psid).

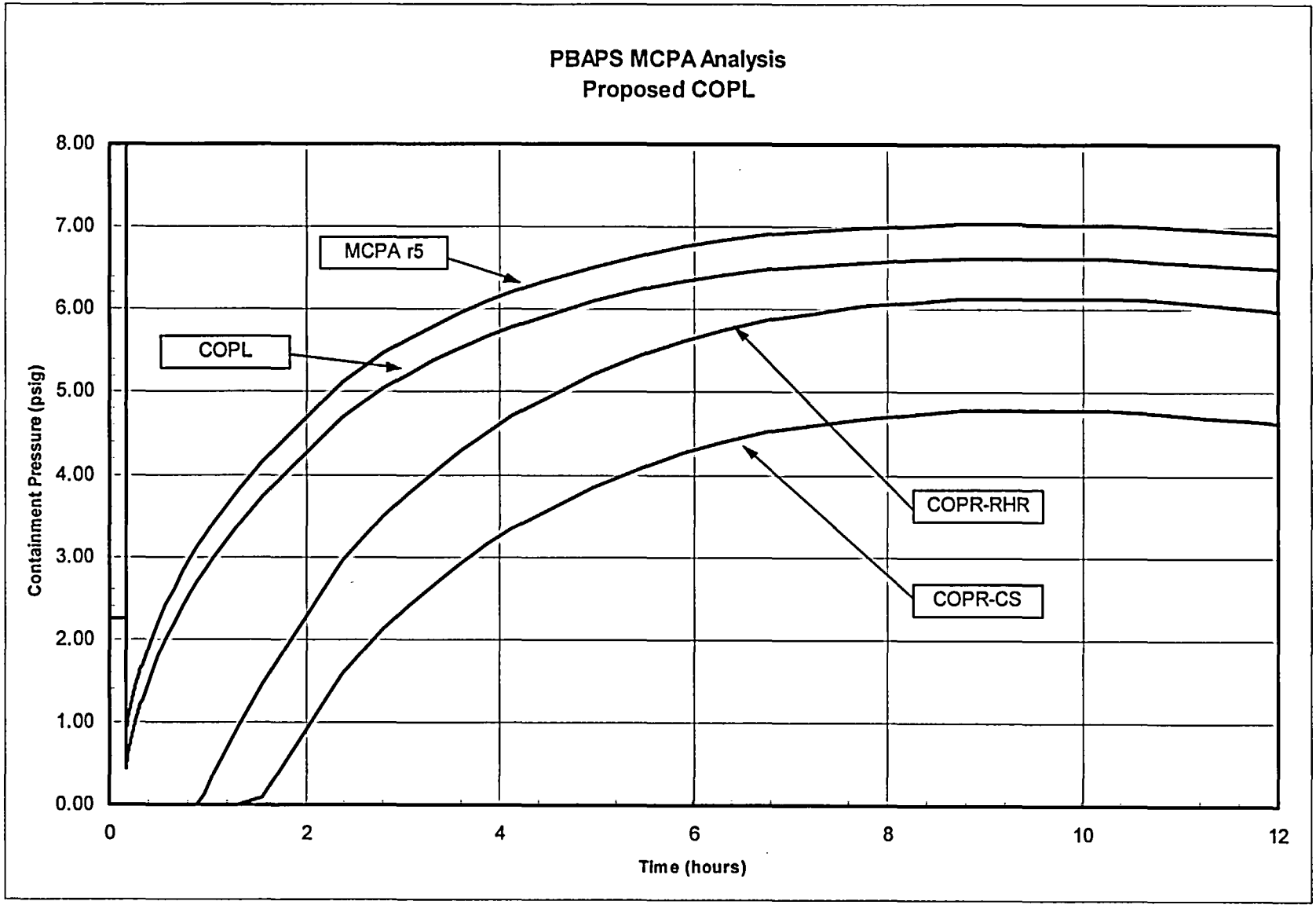
(b) The following table summarizes the COPL, MCPA, and COPR data provided in the attached charts, with a COPL based on a 1 foot margin to the MCPA.

	Peak MCPA (psig)	Peak COPL* (psig)	Peak COPR (RHR, psig)	Peak COPR (CS, psig)	MCPA-COPL (psid)	COPL-COPR (psid)
PM-1013 r3	7.41	6.99	6.11	4.83	0.42	0.88
PM-1013 r5**	7.04	6.62	6.14	4.78	0.42	0.48

* Both COPL values represent a 1 foot margin to the MCPA
** rev. 5 was prepared to address corrections in the calculation write-up of rev. 4 which was prepared for this amendment request.

The Suppression Chamber air space mixing issue will be addressed in question 14.


PBAPS MCPA Analysis
Previous COPL
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
Containment Pressure (psig)
0
2
4
6
8
10
12
Time (hours)
MCPA r5
COPL
COPR-RHR
COPR-CS


PBAPS MCPA Analysis
Proposed COPL
Containment Pressure (psig)
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
0
2
4
6
8
10
12
Time (hours)
MCPA r5
COPL
COPR-RHR
COPR-CS

14. In addressing RG 1.183, Appendix A, LOCA Item 6.1, it is stated in Table B that the radioactive release is mixed with the suppression chamber air space "based on expected steam flow from the drywell to the suppression chamber, even after the initial blowdown."

    a. Is this based on the results of thermal-hydraulic analyses performed for the duration of the release? If so, provide a summary of the analyses for staff review, or

    b. Provide justification for this assumption for the duration of the release.

    RESPONSE:

    The original submittal was based on steam driven exchange through the downcomers. For consistency with the assumptions required for the core activity release (2 hours of no ECCS flow), no steaming is now assumed for the first two hours. The drywell/suppression chamber air space is assumed to be well-mixed thereafter due to flashing associated with core re-flood.

19. Questions regarding the use of the SLC are currently being developed and will be provide in a future RAI.

    RESPONSE: SLC questions identified per the NRC guidance document will be responded to under a separate cover.

20. On Page 15 of Attachment 1 of the submittal, the second paragraph states that Exelon has used the Brockmann-Bixler model for main steamline deposition. The discussion and the data in Table 5 are insufficient to support an NRC staff confirmation. Please provide the following information.

    a. A single-line sketch of the four main steamlines and the isolation valves. Annotate this sketch to identify each of the control volumes assumed by Exelon in the deposition model.

    b. A tabulation of all of the parameters input into the Brockmann-Bixler model for each control volume shown in the sketch (and time step) for which Exelon is crediting deposition. This includes:

        - Flow rate
        - Gas pressure
        - Gas temperature
        - Volume
        - Inner surface area
        - Total pipe bend angle

    c. For each of the bulleted parameters in question 20.b., provide a brief derivation and an explanation of why that assumption is adequately conservative for a design-basis calculation. Address changes in parameters over time, e.g., plant cooldown.

d. Clarify if your analysis addresses a single failure of one of the MSIVs. Such a failure could change the control volume parameters that are input to the deposition model. Previous implementations of main steam deposition have been found acceptable only if the licensee had modeled a limiting single failure. Please explain why Exelon feels that such a limiting failure need not be considered if it is not considered.

e. Since the crediting of main steamline deposition effectively establishes the main steam piping as a fission product mitigation system, the staff expects the piping to meet the requirements of an ESF system, including seismic and single failure considerations. Please confirm that the main steam piping and isolation valves that establish the control volumes for the modeling of deposition were designed and constructed to maintain integrity in the event of the safe shutdown basis earthquake for Peach Bottom. If the design basis for the piping and components does not include integrity during earthquakes, please provide an explanation of how the Peach Bottom design satisfies the prerequisites of the staff-approved NEDC-31858P-A, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control Systems." If piping systems and components at Peach Bottom were previously found by the staff to be seismically rugged using the methodology of this BWROG report, please provide a specific reference to the staff's approval.

RESPONSE:
The use of Brockmann-Bixler approach incorporated in the initial submittal is being abandoned to incorporate a well-mixed methodology to facilitate submittal review, and to credit AEB-98-03 settling velocity treatment.

Furthermore, since deposition in inboard MS piping is being credited, the design basis pipe break is assumed to be a steam line break inside containment in the vicinity of an inboard MSIV.

A single inboard MSIV failure is assumed on the broken line. The MSIV assumed to fail open is the one associated with the broken inboard line that produces the highest dose. To account for possible containment turbulence in the vicinity of the penetration piping, the first two pipe diameters in the penetration will not be credited.

Figure 1 in this attachment, shows a single-line sketch of the four main steam lines and the isolation valves. This figure shows control volumes and break locations.

Table 1 in this attachment, shows all parameters input into the AEB-98-03 based model of the steam lines, and justifies the conservatism, including consideration of plant cool down effects.

For the analysis, leakage is assumed to be distributed evenly between the two worst steam lines. MSIV leakage limits will be 75 scfh maximum per main steam line with a total acceptance criterion of 150 scfh for all four lines.

Only steam line piping that has been seismically qualified is credited in this analysis. PBAPS did not pursue the NEDC-31858P based approach for seismically analyzing and crediting balance of plant equipment, such as turbine shells or the main condenser.

21. On page 53 of Attachment 1 of your submittal, you state that your submittal is in compliance with paragraph 6.3 of Appendix A to RG 1.183, and reference the RADTRAD Brockman-Bixler approach apparently as establishing that conformance. However, paragraph 6.3 of RG 1,183 states that the model should be based on well-mixed volumes, but other models such as slug flow may be used if justified. The Brockman-Bixler model is a slug-flow model. This paragraph did not endorse RADTRAD as an acceptable approach. RG 1.183 states that main steamline deposition will be considered on a case-by-case basis.

    The staff documented its evaluation of the first application of main steamline deposition credit in an AST in Appendix A of NRC staff report, AEB-98-03, "Assessment of the Radiological Consequences for the Perry Pilot Plant Application using the Revised (NUREG-1465) Source Term." The methodology of this report, which can be found online in ADAMS at ML011230531, was used by at least two additional licensees.

    Generally, when the staff has accepted an application of slug flow, the licensee has (1) committed to maintaining a seismically rugged drain path from the 3rd MSIV to and through the condenser, (2) did not assume deposition in piping upstream of the inboard MSIV, (3) assumed a single failure of one of the inboard MSIVs, (4) did not credit a delay time in the onset of the release, and (5) assumed a constant pressure and temperature in the steamline over 30 days. The added conservatism from the above assumptions provided additional margins to compensate for differences in conservatism in slug flow and well-mixed assumptions. Please provide a justification for your proposed modeling approach or re-perform the analyses.

    RESPONSE:

    Reference response to question 20 above. AEB-98-03 methodology is now used to assess aerosol removal in steam line piping. Analogous treatment is used for elemental iodine deposition. No deposition is credited for organic iodine.

    For aerosol settling only horizontal piping is credited, with the lower half providing the settling area. For elemental iodine deposition, all available piping and surface areas are credited.

    Slug flow modeling for MSIV leakage will not be used.

22. Page 13 of Attachment 1 of your submittal provides text that states "an initial 12 hours transport delay is determined." The text suggests that the steamline volume and MSIV leak rate are used to establish this delay. This implies that a delay to fill the steamline is being taken:

    a. Your submittal does not identify this as an alternative to the guidance in RG 1.183. Please explain how this holdup is modeled in the LOCA analysis. Is this modeled as a delay in the onset of the release?

    b. Please explain why this delay assumption is consistent with the assumption of slug flow (Item 6.3, Page 53 of Attachment 1).

    RESPONSE:

    The 12-hour delay would not be consistent with well-mixed modeling, and therefore, will no longer be credited.

23. Based on information provided in your submittal, you have assumed an MSIV leakage rate of 0.62 cfm for the 100 scfh lines, and 0.31 cfm for the 50 scfh line, prior to 24 hours post-accident and reduced values after 24 hours. The staff believes that these values are understated. When the proposed MSIV leakage, in scfh, at test conditions (typically 70 degrees and 25 psig) are scaled to peak containment pressure and temperature (typically 40-50 psig and about 250-350 degrees) the TS leakage past the inboard MSIV has been shown to be 1.3-1.6 cfm, at least double the value you have assumed. However, the temperature of the fluid in the steamlines is based on the steam piping temperatures, typically 500-600 degrees. At the steam piping conditions, the flow in scfm is even higher, typically 4-8 scfm. Please explain the basis of the values you used and why these values are adequately conservative since the effectiveness of deposition decreases with increasing flow.

RESPONSE:

The leakage rates of 0.62 and 0.31 cfm result from correcting the measured outboard 100 scfh flow to inboard pressure conditions. This means applying a factor of (14.7/(14.7+25)), where 25 is the MSIV test pressure and 14.7 is atmospheric pressure, both in psi.

This is consistent (in a reverse direction) with the PBAPS and every other BWR approach to L_a management. For example with current PBAPS limits:

293,900 [cu. ft.] * 0.5 [%/day] * 0.01 [/%] * (14.7 [psia] + 49.1 [psig]) / 14.7 [psia] /24 [hr/day] = 265.7 scfh.

This would be an inboard flow rate of 61.23 cfh. This approach is consistent with 10CFR50, Appendix J and its cited ANSI/ANS Standards. No temperature adjustment is required related to leak rate.

Flow rates in inboard MSIV piping will be the same as the volumetric leak rate, which is, in effect, a mass flow rate from a constant volume. Any heating in inboard MS piping would cause expansion back into containment or the reactor vessel until pressures equalize. However, if outboard piping should be hotter than inboard piping then leakage expansion could be greater than that associated with test conditions. Therefore, for conservatism, the flow rate in outboard piping is adjusted as follows:

75 [scfh] * (550[F] + 460[R]) / (68[F] + 460 [R])

No extrapolation upward from the MSIV test pressure to P_a was deemed necessary as actual pressures only exceed the test pressure for approximately the first 6 minutes of the event.

However, based on a review of the Staff endorsed NEDC-32091 and NEDC-31858P documents, an alternative method of evaluating leak rates is now being applied. This method accounts for partial pressures of water vapor, initial containment non-condensables based on containment response to a Recirculation Suction Line Break, plus H_2 from Zirconium-Water reaction. Use of this methodology for a 100 scfh MSIV leakage acceptance criteria for PBAPS results in a predicted leak rate of 0.58 cfm at containment conditions.

The proposed MSIV leakage limit is now 150 scfh total with a maximum of 75 scfh in any one line. The above method results in a proportionally lower leak rate of 0.437 cfm in the maximum line at containment conditions.

26. Section 12.3.3, "Design Considerations," of the UFSAR states "The main control room, the Technical Support Center (TSC), and the Emergency Operation Facility (EOF) design is based on the airborne fission product inventory in the reactor building following the design-basis LOCA in Unit 2 or 3, using a TID-14844 source term. Shielding and ventilation air treatment are provided such that operators occupying the control room, the TSC, and the EOF and traveling to and from the control room across the site will receive an exposure of less than 5 Rem whole body or its equivalent over the course of the accident." Page 42 of Attachment 1 states "The Technical Support Center at PBAPS is in the Unit 1 Control Room. A review of the current TID-14844-based analysis indicates that it is unnecessary to reanalyze doses therein to assure accessibility. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures are bounded by those previously analyzed." Please provide more details regarding these assessments. Justify the conclusions reached by these qualitative assessments.

RESPONSE:

Doses to personnel in the TSC have been reanalyzed using the same release modeling as used for the control room. Primary differences between the TSC and control room are improved X/Qs at the TSC, and the availability of a recirculation filter, in addition to the intake filter. An unfiltered inleakage allowance of 50% of the filtered intake flow rate is analyzed. Direct shine from external sources such as airborne activity in the unshielded reactor building refuel floor and external cloud are assessed to quantify this contributor.

Table 2 (Attachment 7) provides additional detail on the qualitative and semi-quantitative assessment for the analyzed activities as described in the UFSAR Section 12.3.5.

Further discussion of quantitative and qualitative assessments of control room dose contributors is included in Attachment 6 (Table A, Item 4.2.1). Other vital areas have been reassessed using AST source terms and have been determined to be accessible consistent with their use.

The EOF is located in Coatsville, PA (approximately 30 miles away from the site).

27. Page 49 of Attachment 1, Table B, contains a comparison of the Peach Bottom analysis to Section 4.5 of RG 1.183. The comment column of this table states "However, based on revised containment pressure analysis, the revised TS MSIV leakage is limited to 174 scfh." Proposed insert A (for SR 3.6.1.3.14 on TS page B 3.6-29) states that the total leakage through all four main steamlines must be less than 250 scfh. Please explain this apparent inconsistency.

RESPONSE:

These TS pages are revised to reflect leakage rates that are now limited based on LOCA Dose analyses. (see Attachment 5 to this supplement).

28. Page 52 of Attachment 1, Table B, contains a comparison of the Peach Bottom analysis to Section 6.1 of RG 1.183. The PBAPS analysis column of this table states that it conforms with RG 1.183, but this RG does not endorse mixing between the drywell and the suppression chamber air volume to determine the source term for the MSIV leakage. The assumption that the radioactive release is assumed to instantaneously mix between these two volumes appears to be inconsistent with the timing of the AST.

    a. Is this based on the results of thermal-hydraulic analyses performed for the duration of the release? If so, provide a summary of the analyses for staff review, or

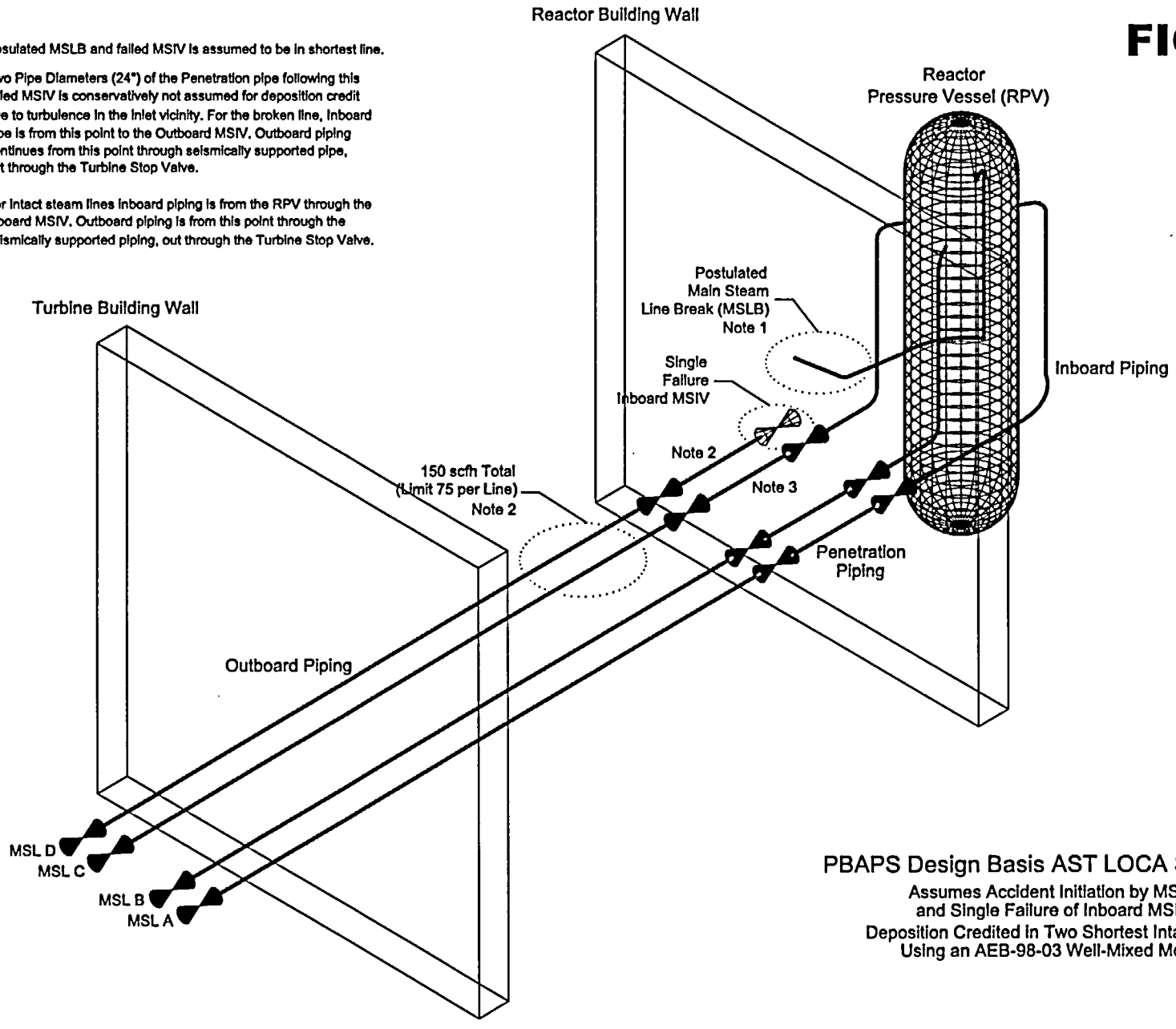
    b. Provide justification for this assumption for the duration of the release.

RESPONSE:
The original submittal was based on steam driven exchange through the downcomers. For consistency with the assumptions required for the core activity release (2 hours of no ECCS flow), no steaming is assumed for the first two hours. The drywell/suppression chamber air space is assumed to be well-mixed thereafter due to flashing associated with core re-flood.

FIGURE 1

Notes:

1. Posulated MSLB and failed MSIV is assumed to be in shortest line.
2. Two Pipe Diameters (24") of the Penetration pipe following this failed MSIV is conservatively not assumed for deposition credit due to turbulence in the inlet vicinity. For the broken line, Inboard pipe is from this point to the Outboard MSIV. Outboard piping continues from this point through seismically supported pipe, out through the Turbine Stop Valve.
3. For intact steam lines inboard piping is from the RPV through the Inboard MSIV. Outboard piping is from this point through the seismically supported piping, out through the Turbine Stop Valve.



PBAPS Design Basis AST LOCA Schematic

Assumes Accident Initiation by MSLB
and Single Failure of Inboard MSIV
Deposition Credited in Two Shortest Intact Lines
Using an AEB-98-03 Well-Mixed Model

PBAPS Unit 3 Determination of MSL Decontamination Factors Due to Iodine Deposition

	Unit 3 Inboard A	Unit 3 Inboard B	Unit 3 Inboard C	Unit 3 [12]Inboard D	Unit 3 Outboard A	Unit 3 Outboard B	Unit 3 Outboard C	Unit 3 [12]Outboard D
[1]Total Pipe Surface Area (ft^2)	687	620	617	115	2101	2036	1971	1618
[1]Total Pipe Volume (ft^3)	338	305	304	56	1034	1002	970	796
[1]Horizontal Total Pipe Surface Area (ft^2)	302	258	255	115	2031	1966	1901	1548
Horizontal Settling Pipe Surface Area (ft^2)	151.19	128.87	127.60	57.58	1015.49	982.98	950.49	774.04
[1]Horizontal Pipe Volume (ft^3)	149	127	126	56	1000	968	936	782
[2]Aerosol Settling Velocity (m/s)	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03	1.170E-03
Aerosol Settling Velocity (ft/s)	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03	3.839E-03
[5,6]Elemental Deposition Velocity 0-24hrs (m/sec)	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06	5.359E-06
[5,6]Elemental Deposition Velocity 24-96hrs (m/sec)	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05	1.248E-05
[5,6]Elemental Deposition Velocity 96-720hrs (m/sec)	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05	7.945E-05
Elemental Deposition Velocity 0-24hrs (ft/sec)	1.758E-05	1.758E-05	1.758E-05	1.758E-05	1.758E-05	1.758E-05	1.758E-05	1.758E-05
Elemental Deposition Velocity 24-96hrs (ft/sec)	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05	4.095E-05
Elemental Deposition Velocity 96-720hrs (ft/sec)	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04	2.607E-04
[5,6]Organic Deposition Velocity 0-24hrs (m/sec)	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09	5.969E-09
[5,6]Organic Deposition Velocity 24-96hrs (m/sec)	1.390E-08	1.390E-08	1.390E-08	1.390E-08	1.390E-08	1.390E-08	1.390E-08	1.390E-08
[5,6]Organic Deposition Velocity 96-720hrs (m/sec)	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08	8.849E-08
Organic Deposition Velocity 0-24hrs (ft/sec)	1.958E-08	1.958E-08	1.958E-08	1.958E-08	1.958E-08	1.958E-08	1.958E-08	1.958E-08
Organic Deposition Velocity 24-96hrs (ft/sec)	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08	4.561E-08
Organic Deposition Velocity 96-720hrs (ft/sec)	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07	2.903E-07
Uncorrected Flow Rate (scfh)	0	0	75	75	0	0	75	75
[11]Artificial Containment Flow Rate 0-2 hrs (cfm)	0.0000	0.0000	0.7736	0.7736	N/A	N/A	N/A	N/A
From Containment Flow Rate 2-24 hrs (cfm)	0.0000	0.0000	0.4375	0.4375	N/A	N/A	N/A	N/A
From Containment Flow Rate 24-720 hrs (cfm)	0.0000	0.0000	0.2187	0.2187	N/A	N/A	N/A	N/A
Pipe Flow Rate 0-24 hrs (cfm)	0.0000	0.0000	0.4375	0.4375	0.0000	0.0000	2.4100	2.4100
Pipe Flow Rate 24-96 hrs (cfm)	0.0000	0.0000	0.2187	0.2187	0.0000	0.0000	1.0298	1.0298
Pipe Flow Rate 96-720 hrs (cfm)	0.0000	0.0000	0.2187	0.2187	0.0000	0.0000	0.7813	0.7813
Pipe Flow Rate 0-24 hrs (cfh)	0.0000	0.0000	26.2489	26.2489	0.0000	0.0000	144.6023	144.6023
Pipe Flow Rate 24-96 hrs (cfh)	0.0000	0.0000	13.1245	13.1245	0.0000	0.0000	61.7898	61.7898
Pipe Flow Rate 24-720 hrs (cfh)	0.0000	0.0000	13.1245	13.1245	0.0000	0.0000	46.8750	46.8750
[3]Aerosol Settling Rate Constant (hr^{-1})	1.40E+01	1.40E+01	1.40E+01	1.41E+01	1.40E+01	1.40E+01	1.40E+01	1.40E+01
[3]Elemental Deposition Rate Constant 0-24hr (hr^{-1})	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01	1.29E-01
[3]Elemental Deposition Rate Constant 24-96hr (hr^{-1})	3.00E-01	3.00E-01	3.00E-01	3.01E-01	3.00E-01	3.00E-01	3.00E-01	3.00E-01
[3]Elemental Deposition Rate Constant 96-720hr (hr^{-1})	1.91E+00	1.91E+00	1.91E+00	1.92E+00	1.91E+00	1.91E+00	1.91E+00	1.91E+00
[3]Organic Deposition Rate Constant 0-24hr (hr^{-1})	1.43E-04	1.43E-04	1.43E-04	1.44E-04	1.43E-04	1.43E-04	1.43E-04	1.43E-04
[3]Organic Deposition Rate Constant 24-96hr (hr^{-1})	3.34E-04	3.34E-04	3.34E-04	3.35E-04	3.34E-04	3.34E-04	3.34E-04	3.34E-04
[3]Organic Deposition Rate Constant 96-720hr (hr^{-1})	2.12E-03	2.12E-03	2.12E-03	2.13E-03	2.12E-03	2.12E-03	2.12E-03	2.12E-03
[4]Aerosol Filter Efficiency (0-24 hrs)	0.00%	0.00%	98.53%	96.81%	0.00%	0.00%	98.91%	98.67%
[4]Aerosol Filter Efficiency (24-96 hrs)	0.00%	0.00%	99.26%	98.38%	0.00%	0.00%	99.53%	99.43%
[4]Aerosol Filter Efficiency (96-720 hrs)	0.00%	0.00%	99.26%	98.38%	0.00%	0.00%	99.64%	99.56%
[4]Elemental Filter Efficiency (0-24 hrs)	0.00%	0.00%	59.82%	21.73%	0.00%	0.00%	46.32%	41.46%
[4]Elemental Filter Efficiency (24-96 hrs)	0.00%	0.00%	87.40%	56.40%	0.00%	0.00%	82.46%	79.43%
[4]Elemental Filter Efficiency (96-720 hrs)	0.00%	0.00%	97.78%	89.17%	0.00%	0.00%	97.53%	97.01%
[4]Organic Filter Efficiency (0-24 hrs)	0.00%	0.00%	0.17%	0.03%	0.00%	0.00%	0.10%	0.08%
[4]Organic Filter Efficiency (24-96 hrs)	0.00%	0.00%	0.77%	0.14%	0.00%	0.00%	0.52%	0.43%
[4]Organic Filter Efficiency (96-720 hrs)	0.00%	0.00%	4.69%	0.91%	0.00%	0.00%	4.21%	3.48%
[7]Pipe Wall Temperature, 0-24hr (F)	558.00							
Pipe Wall Temperature, 0-24hr (K)	565.37							
[7]Pipe Wall Temperature, 24-96hr (F)	410.00							
Pipe Wall Temperature, 24-96hr (K)	483.15							
[7]Pipe Wall Temperature, 96-720hr (F)	200.00							
Pipe Wall Temperature, 96-720hr (K)	366.48							
[8]Condenser Temperature, constant (F)	Not Credited							
Condenser Temperature, constant (K)	Not Credited							
Standard Temperature, constant (F)	68							
[9]MSIV Test Pressure, constant (psig)	25							
[10]Peak P_a Containment Pressure, constant (psig)	49.1							
Atmospheric Pressure, constant (psia)	14.7							
[13]Extrapolation Factor, constant	1.00							

References:

1. Piping Take-off Summary (attached)
2. USNRC Document AEB-98-03, 12/9/1998, Page A-3, Median Value
3. USNRC Document AEB-98-03, 12/9/1998, Page A-2, Formula 2
4. USNRC Document AEB-98-03, 12/9/1998, Page A-2, Formula 4
5. Cline, J.E. "MSIV Leakage Iodine Transport Analysis", 3/26/1991
6. NUREG/CR-6604, RADTRAD Manual, 4/1998; Supplement 1, 6/1999
7. Cline, J.E. 8/1990
8. Condenser Not Credited
9. PBAPS Technical Specification
10. PBAPS Technical Specification
11. Value used to simulate an un-mixed containment volume for the first 2 hours of the LOCA.
12. Main Steam Line Break of longest inboard line assumed for LOCA Deposition credit reduction.
13. No extrapolation from Test pressure to P_a applied due to small differential, short time of that differential, and other conservatisms.

Determination of Inboard MSIV Leak Rates using NEDC-31858P and NEDC-32091 Methodology

Constants											
68	Standard Temperature (°F)										
558	Main Steam Pipe Wall Temp 0-24 hours (°F)										
410	Main Steam Pipe Wall Temp 24-96 hours (°F)										
200	Main Steam Pipe Wall Temp 96-720 hours (°F)										
14.7	Conversion Factor (atm to psi)										
Containment Volumes											
159,000	Drywell Volume (ft^3)										
127,700	Wetwell Volume (ft^3)										
7,200	Reactor Vessel (ft3) space above nominal water level vs. {GE 14,000 ft^3 value}										
293,900	Total Volume (ft^3)										
8322.3663	Total Volume (m^3)										
1.7684	Ratio of Total Volume to Drywell Volume including RPV										
Containment Temperatures and Pressures per Containment Analysis for RSLB in PM-1061, R0											
276	DW Temp (°F) at minimum DW-WW differential (at ~ 69 seconds)										
131	WW Temp (°F) at minimum DW-WW differential (at ~ 69 seconds)										
213.0	Average Bulk Temperature (°F)										
46.1	DW Pressure (psia) {use for pressure vessel as well}										
43.9	WW Pressure (psia)										
45.1	Average Bulk Pressure (psia)										
3.07	Average Bulk Pressure (atmospheres)										
Hydrogen Contribution from Zirconium Water Reaction											
764	assemblies					{PBAPS Value}					
102.00	lbs Zr/assembly					{NEDC-31858P}					
7.87	cubic feet H_2 per lb Zr					{NEDC-31858P}					
0.20	fraction of Zr undergoing metal water reaction					{NEDC-31858P}					
122658.67	Total Hydrogen (ft^3)					{Calculated PBAPS Value}					
167782.42	Corrected to bulk average temperature					{Calculated PBAPS Value}					
0.5708827	Partial Pressure of Hydrogen (atmospheres)					{Calculated PBAPS Value}					
3.64	Total {H_2, N_2, H_2O} Pressure (atmospheres)					{Calculated PBAPS Value}					
Inboard Leak Rate Determination per NEDC-32091, Section B.1.3, Duane Arnold Example based.											
A	B	C	D								
0	0	75	75	Containment Leak Rate (scfh) {use as basis for outboard flow rate}							
0	0	0.21435	0.21435	Leak Rate in %/day							
0.0000	0.0000	0.4375	0.4375	Inboard Leak Flow Rate (cfm)							
0.0000	0.0000	26.2489	26.2489	Inboard Leak Flow Rate (cfh)							
Note that no extrapolation from test pressure to Pa is required based on the NEDC-31858P note											
that these containment conditions are essentially equivalent to test conditions.											

Table 1: Main Steam Line Deposition Parameters - Page 2 of 7

Main Steam Piping Summary						
23.624	Main Steam 24 inch pipe ID					
A	B	C	D			
PBAPS Unit 2						
Nodalization (Horizontals)						
296	254	254	300	Node 1 Surface Area (sq. ft.)		
146	125	125	148	Node 1 Volume (cu. ft.)		
153	140	140	153	Node 2 Surface Area (sq. ft.)		
75	69	69	75	Node 2 Volume (cu. ft.)		
1794	1838	1882	1927	Node 3 Surface Area (sq. ft.)		
883	905	926	948	Node 3 Volume (cu. ft.)		
Nodalization (Totals)						
667	616	616	671	Node 1 Surface Area (sq. ft.)		
328	303	303	330	Node 1 Volume (cu. ft.)		
153	140	140	153	Node 2 Surface Area (sq. ft.)		
75	69	69	75	Node 2 Volume (cu. ft.)		
1863	1907	1952	1997	Node 3 Surface Area (sq. ft.)		
917	939	961	983	Node 3 Volume (cu. ft.)		
A	B	C	D			
PBAPS Unit 3						
Nodalization (Horizontals)						
302	258	255	307	Node 1 Surface Area (sq. ft.)		
149	127	126	151	Node 1 Volume (cu. ft.)		
140	140	140	140	Node 2 Surface Area (sq. ft.)		
69	69	69	69	Node 2 Volume (cu. ft.)		
1891	1826	1761	1548	Node 3 Surface Area (sq. ft.)		
931	899	867	762	Node 3 Volume (cu. ft.)		
Nodalization (Totals)						
687	620	617	685	Node 1 Surface Area (sq. ft.)		
338	305	304	337	Node 1 Volume (cu. ft.)		
140	140	140	140	Node 2 Surface Area (sq. ft.)		
69	69	69	69	Node 2 Volume (cu. ft.)		
1961	1896	1831	1618	Node 3 Surface Area (sq. ft.)		
965	933	901	796	Node 3 Volume (cu. ft.)		

PBAPS Unit 3 Main Steam Line A

Inner Diameter (in.)= 23.624

Location	Horizontal	Volume (ft^3)	Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard	TRUE	4.81	9.77	4.81	9.77
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	100.55	204.30	0	0.00
Inboard	FALSE	39.52	80.30	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	83.98	170.63	83.98	170.63
Inboard	TRUE	9.896	20.11	9.896	20.11
Inboard	FALSE	49.15	99.86	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	19.03	38.67	19.03	38.67
Penetration	TRUE	68.85	139.89	68.85	139.89
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	FALSE	34.26	69.61	0	0.00
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	120.28	244.39	120.28	244.39
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	424.02	861.54	424.02	861.54
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	241.83	491.36	241.83	491.36
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	92.75	188.45	92.75	188.45
	Totals	1371.886	2787.44	1148.41	2333.37

	Total Volume (ft^3)	Total Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard (Node 1)	338.05	686.85	148.83	302.39
Penetration (Node 2)	68.85	139.89	68.85	139.89
Outboard (Node 3)	964.99	1960.70	930.73	1891.09
Totals	1371.89	2787.44	1148.41	2333.37

PBAPS Unit 3 Main Steam Line B

Inner Diameter (in.)= 23.624

Location	Horizontal	Volume (ft^3)	Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard	TRUE	5.45	11.07	5.45	11.07
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	100.03	203.24	0	0.00
Inboard	FALSE	35.57	72.27	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	47.11	95.72	47.11	95.72
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	42.69	86.74	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	22.44	45.59	22.44	45.59
Penetration	TRUE	68.85	139.89	68.85	139.89
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	FALSE	34.5	70.10	0	0.00
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	131.94	268.08	131.94	268.08
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	424.02	861.54	424.02	861.54
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	210.35	427.40	210.35	427.40
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	80.57	163.70	80.57	163.70
	Totals	1307.22	2656.05	1094.43	2223.70

	Total Volume (ft^3)	Total Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard (Node 1)	305.14	619.99	126.85	257.74
Penetration (Node 2)	68.85	139.89	68.85	139.89
Outboard (Node 3)	933.23	1896.17	898.73	1826.07
Totals	1307.22	2656.05	1094.43	2223.70

PBAPS Unit 3 Main Steam Line C

Inner Diameter (in.)= 23.624

Location	Horizontal	Volume (ft^3)	Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard	TRUE	5.45	11.07	5.45	11.07
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	100.03	203.24	0	0.00
Inboard	FALSE	35.57	72.27	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	47.11	95.72	47.11	95.72
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	42.69	86.74	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	21.19	43.05	21.19	43.05
Penetration	TRUE	68.85	139.89	68.85	139.89
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	FALSE	34.5	70.10	0	0.00
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	143.6	291.77	143.6	291.77
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	424.02	861.54	424.02	861.54
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	178.89	363.47	178.89	363.47
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	68.39	138.96	68.39	138.96
	Totals	1273.99	2588.53	1061.20	2156.18

	Total Volume (ft^3)	Total Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard (Node 1)	303.89	617.45	125.60	255.20
Penetration (Node 2)	68.85	139.89	68.85	139.89
Outboard (Node 3)	901.25	1831.19	866.75	1761.09
Totals	1273.99	2588.53	1061.20	2156.18

Table 1: Main Steam Line Deposition Parameters - Page 6 of 7

PBAPS Unit 3 Main Steam Line D

Inner Diameter (in.)= 23.624

Location	Horizontal	Volume (ft^3)	Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard	TRUE	4.32	8.78	4.32	8.78
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	FALSE	100.55	204.30	0	0.00
Inboard	FALSE	39.52	80.30	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	83.98	170.63	83.98	170.63
Inboard	TRUE	12.424	25.24	12.424	25.24
Inboard	FALSE	46.44	94.36	0	0.00
Inboard	TRUE	10.37	21.07	10.37	21.07
Inboard	TRUE	19.03	38.67	19.03	38.67
Penetration	TRUE	68.85	139.89	68.85	139.89
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	FALSE	34.56	70.22	0	0.00
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	155.29	315.52	155.29	315.52
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	351.12	713.42	351.12	713.42
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	147.44	299.57	147.44	299.57
Outboard	TRUE	10.37	21.07	10.37	21.07
Outboard	TRUE	56.21	114.21	56.21	114.21
	Totals	1202.694	2443.67	981.62	1994.50

	Total Volume (ft^3)	Total Surface Area (ft^2)	Horizontal Volume (ft^3)	Horizontal Surface Area (ft^2)
Inboard (Node 1)	337.37	685.49	150.86	306.53
Penetration (Node 2)	68.85	139.89	68.85	139.89
Outboard (Node 3)	796.47	1618.29	761.91	1548.07
Totals	1202.69	2443.67	981.62	1994.50

ATTACHMENT 2

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Revised TS Pages Markups and TS Markup Inserts pages

<u>UNITS 2 & 3</u>
Inserts Page
3.6-16
5.0-13

TS Inserts For PBAPS AST LAR

Insert 1

or Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," 1989.

Insert 2

Verify combined MSIV leakage rate for all four main steam lines is ≤ 150 scfh, and ≤ 75 scfh for any one steam line, when tested at ≥ 25 psig.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ INSERT 2	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section~~s 5 (SGT System only) and~~ 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
MCREV System	2700 to 3300

c. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	~~ESF Ventilation System~~	
	~~SGT System~~	MCREV System
Penetration (%)	~~5~~	5
Face Velocity (FPM)	~~60~~	57
Relative Humidity: (%)	~~70~~	95

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	~~Verify leakage rate through each MSIV is ≤ 11.5 scfh when tested at ≥ 25 psig.~~ INSERT 2	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system~~s~~ that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section~~s 6 (SGT System only) and~~ 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
~~SGT System~~	~~7200 to 8800~~
MCREV System	2700 to 3300

c. Demonstrate for ~~each of~~ the ~~ESF~~ MCREV system~~s~~ that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	~~ESF Ventilation System~~	
	~~SGT System~~	MCREV System
Penetration (%)	~~5~~	5
Face Velocity (FPM)	~~60~~	57
Relative Humidity: (%)	~~70~~	95

(continued)

ATTACHMENT 3

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Revised TS Bases Inserts
(For information only)

UNITS 2 & 3

Revised TS Bases Inserts
B3.6-29

PBAPS Units 2 and 3 Technical Specification Bases Markup Inserts

INSERT A {pg. B 3.6-29}

Total leakage through all four main steam lines must be ≤ 150 scfh, and ≤ 75 scfh for any one steam line, when tested at ≥ 25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

INSERT B {pg. B 3.1-39}

The SLC System is also used to maintain suppression pool pH at or above 7 following a loss of coolant accident (LOCA) involving significant fission product releases. Maintaining suppression pool pH levels at or above 7 following an accident ensures that iodine will be retained in the suppression pool water.

INSERT C {pg. B 3.1-41}

In MODES 1, 2, and 3, the SLC System must be OPERABLE to ensure that offsite doses remain within 10 CFR 50.67 (Ref. 3) limits following a LOCA involving significant fission product releases. The SLC System is designed to maintain suppression pool pH at or above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water.

INSERT D {pg. B 3.3-156}

. Both channels are also required to be OPERABLE in MODES 1, 2, and 3, since the SLC System is also designed to maintain suppression pool pH above 7 following a LOCA to ensure that iodine will be retained in the suppression pool water. These

INSERT E {pg. B 3.6-73}

The function of the secondary containment is to receive fission products that may leak from primary containment or from systems in secondary containment following a Design Basis Accident (DBA) and, in conjunction with the Standby Gas Treatment System (SGT) and closure of certain valves whose lines penetrate the secondary containment, to provide for elevated release through the Main Stack.

INSERT F {pg. B 3.6-76}

The SGT System exhausts the secondary containment atmosphere to the environment through the elevated release point provided by the Main Stack.

To ensure that this exhaust pathway is used, SR 3.6.4.1.3

INSERT G {pg. B 3.6-85}

The primary function of the SGT System is to ensure that radioactive materials that leak from primary containment into the secondary containment following a Design Basis Accident (DBA) are discharged through the elevated release provided by the Main Stack.

INSERT H {pg. B 3.6-85}

These filters are not credited in any DBA analysis.

INSERT I {pg. B 3.6-86}

The design basis for the SGT System is to mitigate the consequences of a loss of coolant accident by providing a controlled, elevated release path. The SGT system also provides this function for OPDRVs. For all events where required, the SGT System automatically initiates to reduce, via an elevated release, the consequences of radioactive material released to the environment.

The HEPA filter and charcoal adsorber provided in the SGT System are not credited for any DBA analysis.

INSERT J {pg. B 3.6-90}

The only credited safety function of the SGT System is to provide a secondary containment vacuum sufficient to assure that discharges from the secondary containment will be through the Main Stack. The VFTP test 5.5.7.d. provides verification that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers is acceptable. SR 3.6.4.1.3 and SR 3.6.4.1.4 provide assurance that sufficient vacuum in the secondary containment is established with the time period as used in the DBA LOCA analysis.

INSERT K {pg. B 3.7-16}

Additionally, the MCREV System is designed to maintain the control room environment for a 30-day occupancy after a DBA without exceeding 5 rem TEDE.

INSERT L {pg. B 3.7-16}

The MCREV System is credited as operating following a loss of coolant accident. The MCREV System is not credited in the analysis of the fuel handling accident, the main steam line break, or the control rod drop accident,

INSERT M {pg B 3.6-74}

Secondary containment is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT N {pg B 3.6-87}

The SGT System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT P {pg B 3.6-79}

SCIVs are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT Q {pg B 3.8-40}

involving recently irradiated fuel. With respect to moving irradiated fuel assemblies, AC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours)

INSERT R {PG B 3.8-42, 43, 72, 73, 74, 94, and 95}

involving recently irradiated fuel

INSERT S {pg B 3.8-94}

With respect to moving irradiated fuel assemblies, AC and DC electrical power are only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT T {pg B 3.8-74}

With respect to moving irradiated fuel assemblies, DC electrical power is only required to mitigate fuel handling accidents involving recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT U {pg B 3.6-75, 3.6-82, 3.6-88, 3.6-89, 3.7-18, 3.7-19}

, since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

INSERT V {pg B 3.8-44, 74}

The Required Actions are modified by a Note indicating that LCO 3.0.3 does not apply since the movement of recently irradiated fuel can only be performed in MODES 4 and 5.

INSERT W {pg B 3.3-174}

The Functions are only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT X {pg B 3.3-182}

The MCREV System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT Y {pg B 3.1-40}

The sodium pentaborate solution in the SLC System is also used, post-LOCA, to maintain ECCS fluid pH above 7. The system parameters used in the calculation are the Boron-10 minimum mass of 162.7 lbm, and an upper bound Boron-10 enrichment of 65%.

INSERT Z {pg B 3.7-17}

The MCREV System is only required to be OPERABLE during handling of recently irradiated fuel (i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

INSERT AA {pg B 3.8-22, 3.8-38, 3.8-70}

(i.e., fuel that has occupied part of a critical reactor core within the previous 24 hours).

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~

INSERT A

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

~~Leakage through each MSIV must be ≤ 11.5 scfh when tested at ≥ P_t (25 psig). The analyses in Reference 1 are based on treatment of MSIV leakage as a secondary containment bypass leakage, independent of a primary to secondary containment leakage analyzed at 1.27 L_a. In the Reference 1 analysis all 4 steam lines are assumed to leak at the TS Limit. This ensures that MSIV leakage is properly accounted for in determining the overall impacts of primary containment leakage. The Frequency is required by the Primary Containment Leakage Rate Testing Program.~~

INSERT A

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

ATTACHMENT 4

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Revised Camera-ready TS pages

<u>UNITS 2 & 3</u>
3.6-16
5.0-13

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	Verify combined MSIV leakage rate for all four main steam lines is ≤ 150 scfh, and ≤ 75 scfh for any one steam line, when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for the MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
MCREV System	2700 to 3300

c. Demonstrate for the MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	MCREV System
Penetration (%)	5
Face Velocity (FPM)	57
Relative Humidity: (%)	95

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.1.3.14	Verify combined MSIV leakage rate for all four main steam lines is ≤ 150 scfh, and ≤ 75 scfh for any one steam line, when tested at ≥ 25 psig.	In accordance with the Primary Containment Leakage Rate Testing Program
SR 3.6.1.3.15	Verify each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is blocked to restrict opening greater than the required maximum opening angle.	24 months
SR 3.6.1.3.16	Replace the inflatable seal of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve.	96 months

5.5 Programs and Manuals

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for the MCREV system that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested in accordance with Regulatory Guide 1.52, Revision 2, Section 5d, and ASME N510-1989, Section 11, at the system flowrate specified below.

ESF Ventilation System	Flowrate (cfm)
MCREV System	2700 to 3300

c. Demonstrate for the MCREV system that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, Section 6b, shows the methyl iodide penetration less than the value specified below when tested in accordance with the laboratory testing criteria of ASTM D3803-1989 at a temperature of 30 degrees C [86 degrees F], face velocity, and the relative humidity specified below.

	MCREV System
Penetration (%)	5
Face Velocity (FPM)	57
Relative Humidity: (%)	95

(continued)

ATTACHMENT 5

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Revised Camera-ready TS Bases Pages

(For information only)

UNITS 2 & 3

B3.6-29

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

Total leakage through all four main steam lines must be ≤150 scfh, and ≤75 scfh for any one steam line, when tested at ≥25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.3.13

This SR ensures that in case the non-safety grade instrument air system is unavailable, the SGIG System will perform its design function to supply nitrogen gas at the required pressure for valve operators and valve seals supported by the SGIG System. The 24 month Frequency was developed considering it is prudent that this Surveillance be performed only during a plant outage. Operating experience has shown that these components will usually pass this Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.1.3.14

Total leakage through all four main steam lines must be ≤150 scfh, and ≤75 scfh for any one steam line, when tested at ≥25 psig. The analysis in Reference 1 is based on treatment of MSIV leakage as secondary containment bypass leakage, independent of the primary to secondary containment leakage analyzed at L_a. The Frequency is in accordance with the Primary Containment Leakage Rate Testing Program.

SR 3.6.1.3.15

Verifying the opening of each 6 inch and 18 inch primary containment purge valve and each 18 inch primary containment exhaust valve is restricted by a blocking device to less than or equal to the required maximum opening angle specified in the UFSAR (Ref. 4) is required to ensure that the valves can close under DBA conditions within the times in the analysis of Reference 1. If a LOCA occurs, the purge and exhaust valves must close to maintain primary containment leakage within the values assumed in the accident analysis. At other times pressurization concerns are not present, thus the purge and exhaust valves can be fully open. The 24 month Frequency is appropriate because the blocking devices may be removed during a refueling outage.

(continued)

ATTACHMENT 6

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Regulatory Guide 1.183 Comparison Table Revision

REGULATORY GUIDE 1.183 COMPARISON

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
3.1	The inventory of fission products in the reactor core and available for release to the containment should be based on the maximum full power operation of the core with, as a minimum, current licensed values for fuel enrichment, fuel burnup, and an assumed core power equal to the current licensed rated thermal power times the ECCS evaluation uncertainty. The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code such as ORIGEN 2 or ORIGEN-ARP. Core inventory factors (Ci/MWt) provided in TID 14844 and used in some analysis computer codes were derived for low burnup, low enrichment fuel and should not be used with higher burnup and higher enrichment fuels.	Conforms	ORIGEN 2.1 based methodology was used to determine core inventory. Power level used was 3514.9 MWt which is approximately the current licensed reactor thermal power or 3514 MWt. These source terms were evaluated at end-of-cycle and at beginning of cycle (100 effective full power days (EFPD) to achieve equilibrium) conditions and worst case inventory used for the selected isotopes. These values were then divided by 3514.9 MWt to obtain activity in units of Ci/MWt. Accident analyses are based on a 3528 MWt power level that is the current accident analysis design basis allowance for instrument uncertainty. Source terms are based on a 2 year fuel cycle with a nominal 711 EFPD per cycle.
3.1	For the DBA LOCA, all fuel assemblies in the core are assumed to be affected and the core average inventory should be used. For DBA events that do not involve the entire core, the fission product inventory of each of the damaged fuel rods is determined by dividing the total core inventory by the number of fuel rods in the core. To account for differences in power level across the core, radial peaking factors from the facility's core operating limits report (COLR) or technical specifications should be applied in determining the inventory of the damaged rods.	Conforms	Peaking factors of 1.7 are used for DBA events that do not involve the entire core, with fission product inventories for damaged fuel rods determined by dividing the total core inventory by the number of fuel rods in the core.
3.1	No adjustment to the fission product inventory should be made for events postulated to occur during power operations at less than full rated power or those postulated to occur at the beginning of core life. For events postulated to occur while the facility is shutdown, e.g., a fuel handling accident, radioactive decay from the time of	Conforms	No adjustments for less than full power are made in any analyses.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections

RG Section	RG Position	PBAPS Analysis	Comments
	shutdown may be modeled.		
3.2	The core inventory release fractions, by radionuclide groups, for the gap release and early in-vessel damage phases for DBA LOCAs are listed in Table 1 for BWRs and Table 2 for PWRs. These fractions are applied to the equilibrium core inventory described in Regulatory Position 3.1. **Table 1** **BWR Core Inventory Fraction Released Into Containment** **Group** / **Gap Release Phase** / **Early In-Vessel Phase** / **Total** Noble Gases 0.05 0.95 1.0 Halogens 0.05 0.25 0.3 Alkali Metals 0.05 0.20 0.25 Tellurium Metals 0.00 0.05 0.05 Ba, Sr 0.00 0.02 0.02 Noble Metals 0.00 0.0025 0.0025 Cerium Group 0.00 0.0005 0.0005 Lanthanides 0.00 0.0002 0.0002 Footnote 10: *The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak rod burnup up to 62,000 MWD/MTU. The data in this section may not be applicable to cores containing mixed oxide (MOX) fuel.*	Conforms	The fractions from Regulatory Position 3.1, Table 1 are used. Footnote 10 criteria are met.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
3.2	For non-LOCA events, the fractions of the core inventory assumed to be in the gap for the various radionuclides are given in Table 3. The release fractions from Table 3 are used in conjunction with the fission product inventory calculated with the maximum core radial peaking factor. **Table 3** **Non-LOCA Fraction of Fission Product Inventory in Gap** **Group** — **Fraction** I-131 — 0.08 Kr-85 — 0.10 Other Noble Gases — 0.05 Other Halogens — 0.05 Alkali Metals — 0.12 Footnote 11: *The release fractions listed here have been determined to be acceptable for use with currently approved LWR fuel with a peak burnup up to 62,000 MWD/MTU provided that the maximum linear heat generation rate does not exceed 6.3 kw/ft peak rod average power for rods with burnups that exceed 54 GWD/MTU. As an alternative, fission gas release calculations performed using NRC-approved methodologies may be considered on a case-by-case basis. To be acceptable, these calculations must use a projected power history that will bound the limiting projected plant-specific power history for the specific fuel load. For the BWR rod drop accident and the PWR rod ejection accident, the gap fractions are assumed to be 10% for iodines and noble gases.*	Conforms	Complies with Note 11 of Table 3. Peaking factor of 1.7 used for DBA events that do not involve the entire core.
3.3	Table 4 tabulates the onset and duration of each sequential release phase for DBA LOCAs at PWRs and BWRs. The specified onset is the time following the initiation of the accident (i.e., time = 0). The early in-vessel phase immediately follows the gap release phase. The activity released from the core during each release phase should be modeled as increasing in a linear fashion over the	Conforms	The BWR durations from Table 4 are used. LOCA is modeled in a linear fashion. Non-LOCA is modeled as an instantaneous release.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	duration of the phase. For non-LOCA DBAs, in which fuel damage is projected, the release from the fuel gap and the fuel pellet should be assumed to occur instantaneously with the onset of the projected damage. **Table 4** **LOCA Release Phases** **PWRs** **BWRs** **Phase** **Onset** **Duration** **Onset** **Duration** Gap Release 30 sec 0.5 hr 2 min 0.5 hr Early In-Vessel 0.5 hr 1.3 hr 0.5 hr 1.5 hr		
3.3	For facilities licensed with leak-before-break methodology, the onset of the gap release phase may be assumed to be 10 minutes. A licensee may propose an alternative time for the onset of the gap release phase, based on facility-specific calculations using suitable analysis codes or on an accepted topical report shown to be applicable for the specific facility. In the absence of approved alternatives, the gap release phase onsets in Table 4 should be used.	Not Applicable	PBAPS does not use leak-before-break methodology for DBA analyses.
3.4	Table 5 lists the elements in each radionuclide group that should be considered in design basis analyses. **Table 5** **Radionuclide Groups** **Group** **Elements** Noble Gases Xe, Kr Halogens I, Br Alkali Metals Cs, Rb Tellurium Group Te, Sb, Se, Ba, Sr Noble Metals Ru, Rh, Pd, Mo, Tc, Co Lanthanides La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am Cerium Ce, Pu, Np	Conforms	The nuclides used are the 60 identified as being potentially important dose contributors to total effective dose equivalent (TEDE) in the RADTRAD code, which encompasses those listed in RG 1.183, Table 5.
3.5	Of the radioiodine released from the reactor coolant system (RCS) to the containment in a postulated accident, 95 percent of the iodine	Conforms	This guidance is applied in the analyses.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	released should be assumed to be cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. This includes releases from the gap and the fuel pellets. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form. The same chemical form is assumed in releases from fuel pins in FHAs and from releases from the fuel pins through the RCS in DBAs other than FHAs or LOCAs. However, the transport of these iodine species following release from the fuel may affect these assumed fractions. The accident-specific appendices to this regulatory guide provide additional details.		
3.6	The amount of fuel damage caused by non-LOCA design basis events should be analyzed to determine, for the case resulting in the highest radioactivity release, the fraction of the fuel that reaches or exceeds the initiation temperature of fuel melt and the fraction of fuel elements for which the fuel clad is breached. Although the NRC staff has traditionally relied upon the departure from nucleate boiling ratio (DNBR) as a fuel damage criterion, licensees may propose other methods to the NRC staff, such as those based upon enthalpy deposition, for estimating fuel damage for the purpose of establishing radioactivity releases.	Conforms	Fuel damage assessment for CRDA and FHA are based on GESTAR standard analyses for GE14 fuel.
4.1.1	The dose calculations should determine the TEDE. TEDE is the sum of the committed effective dose equivalent (CEDE) from inhalation and the deep dose equivalent (DDE) from external exposure. The calculation of these two components of the TEDE should consider all radionuclides, including progeny from the decay of parent radionuclides that are significant with regard to dose consequences and the released radioactivity.	Conforms	TEDE is calculated, with significant progeny included.
4.1.2	The exposure-to-CEDE factors for inhalation of radioactive material should be derived from the data provided in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" (Ref. 19). Table 2.1 of Federal Guidance Report 11, "Limiting Values of	Conforms	Federal Guidance Report 11 dose conversion factors (DCFs) are used.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion" (Ref. 20), provides tables of conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the CEDE.		
4.1.3	For the first 8 hours, the breathing rate of persons offsite should be assumed to be 3.5×10^{-4} cubic meters per second. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.8×10^{-4} cubic meters per second. After that and until the end of the accident, the rate should be assumed to be 2.3×10^{-4} cubic meters per second.	Conforms	The analysis uses values to three significant figures that correspond to the rounded values in Section 4.1.3 of RG 1.183.
4.1.4	The DDE should be calculated assuming submergence in semi-infinite cloud assumptions with appropriate credit for attenuation by body tissue. The DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose to the TEDE. Table III.1 of Federal Guidance Report 12, "External Exposure to Radionuclides in Air, Water, and Soil" (Ref. 21), provides external EDE conversion factors acceptable to the NRC staff. The factors in the column headed "effective" yield doses corresponding to the EDE.	Conforms	Federal Guidance Report 12 conversion factors are used.
4.1.5	The TEDE should be determined for the most limiting person at the EAB. The maximum EAB TEDE for any two-hour period following the start of the radioactivity release should be determined and used in determining compliance with the dose criteria in 10 CFR 50.67. The maximum two-hour TEDE should be determined by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments for successive two-hour periods. The maximum TEDE obtained is submitted. The time increments should appropriately reflect the progression of the accident to capture the peak dose interval between the start of the	Conforms	The maximum two-hour LOCA EAB dose is as follows: PC Leakage: **0.0 to 2.0 hours (6.907 Rem** TEDE) due to the 15-minute unfiltered, ground-level SC drawdown time. MSIV Leakage: **11.8 to 13.8 hours (1.155** Rem TEDE). ECCS Leakage: **2.0 to 4.0 hours (0.104** Rem TEDE)

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	event and the end of radioactivity release (see also Table 6). Footnote 14: *With regard to the EAB TEDE, the maximum two-hour value is the basis for screening and evaluation under 10 CFR 50.59. Changes to doses outside of the two-hour window are only considered in the context of their impact on the maximum two-hour EAB TEDE.*		Conservatively, the maximum 2-hour period dose was determined by adding the maximum 2-hour dose for each of the components listed above even though they do not occur simultaneously. This yields: 6.907 + 1.155 + 0.104 = 8.166 Rem TEDE (Rounded up to 8.2 Rem TEDE).
4.1.6	TEDE should be determined for the most limiting receptor at the outer boundary of the low population zone (LPZ) and should be used in determining compliance with the dose criteria in 10 CFR 50.67.	Conforms	This guidance is applied in the analyses.
4.1.7	No correction should be made for depletion of the effluent plume by deposition on the ground.	Conforms	No such corrections made in the analyses.
4.2.1	The TEDE analysis should consider all sources of radiation that will cause exposure to control room personnel. The applicable sources will vary from facility to facility, but typically will include: Contamination of the control room atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility, Contamination of the control room atmosphere by the intake or infiltration of airborne radioactive material from areas and structures adjacent to the control room envelope, Radiation shine from the external radioactive plume released from the facility, Radiation shine from radioactive material in the reactor containment, Radiation shine from radioactive material in systems and components inside or external to the control room envelope, e.g., radioactive material buildup in recirculation filters.	Conforms	The principal source of dose within the control room is due to airborne activity. Calculations of doses from reactor building airborne activity have been recalculated with AST source term assumptions, no credit for contained structures except floors and exterior walls, and with a relatively detailed geometry treatment. SGTS and MCREV filters are well away and/or shielded from the Control Room and have not historically been considered a source for operator doses. This historical conclusion continues to apply as discussed below. **Gamma shine from reactor building:** This component includes shine from the unshielded refuel floor and from the shielded

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
			volume below the refuel floor. Reanalysis with AST source terms indicate a dose contribution of only 0.05 Rem for the duration of the accident. **SGTS filter shine:** This effect due to this source is negligible because the SGTS filter assembly is located on the 91'6" elevation of the radwaste building. The Control Room is located on the 165' elevation within the turbine building, well away from the filters. **MCREV filter shine:** The effect due to the MCREV filters is negligible because the filters are 30' away from the control room air space with an intervening 2' concrete shield. This conclusion is based on experience from other Exelon units with similar geometry. **Primary containment shine:** The 2' reactor building wall plus the 5' containment wall provides ample shielding for the control room. **External cloud:** The control room in an interior space, surrounded by its own 2' thick wall and ceiling concrete shielding. Therefore, doses due to the external cloud is negligible.
4.2.2	The radioactive material releases and radiation levels used in the control room dose analysis should be determined using the same	Conforms	The source term, transport, and release methodology is the same for both the control

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	source term, transport, and release assumptions used for determining the EAB and the LPZ TEDE values, unless these assumptions would result in non-conservative results for the control room.		room and offsite locations.
4.2.3	The models used to transport radioactive material into and through the control room, and the shielding models used to determine radiation dose rates from external sources, should be structured to provide suitably conservative estimates of the exposure to control room personnel.	Conforms	This guidance is applied in the analyses.
4.2.4	Credit for engineered safety features that mitigate airborne radioactive material within the control room may be assumed. Such features may include control room isolation or pressurization, or intake or recirculation filtration. Refer to Section 6.5.1, "ESF Atmospheric Cleanup System," of the SRP (Ref. 3) and Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for Post-accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" (Ref. 25), for guidance.	Conforms	Control Room pressurization and intake filtration are credited in the LOCA accident analysis. No credit is taken in the FHA, MSLB and CRDA accident analyses. No credit is taken for SGTS HEPA or charcoal adsorber filtration in any accident.
4.2.5	Credit should generally not be taken for the use of personal protective equipment or prophylactic drugs. Deviations may be considered on a case-by-case basis.	Conforms	Such credits are not taken.
4.2.6	The dose receptor for these analyses is the hypothetical maximum exposed individual who is present in the control room for 100% of the time during the first 24 hours after the event, 60% of the time between 1 and 4 days, and 40% of the time from 4 days to 30 days. For the duration of the event, the breathing rate of this individual should be assumed to be 3.5×10^{-4} cubic meters per second.	Conforms	The cited occupancy factors and breathing rate are used. An unrounded breathing rate of 3.47E-04 m^3/sec is used.

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
4.2.7	Control room doses should be calculated using dose conversion factors identified in Regulatory Position 4.1 above for use in offsite dose analyses. The DDE from photons may be corrected for the difference between finite cloud geometry in the control room and the semi-infinite cloud assumption used in calculating the dose conversion factors. The following expression may be used to correct the semi-infinite cloud dose, DDE_{∞}, to a finite cloud dose, DDE_{finite}, where the control room is modeled as a hemisphere that has a volume, V, in cubic feet, equivalent to that of the control room (Ref. 22). $DDE_{finite} = \frac{DDE_{\infty} V^{0.338}}{1173}$	Conforms	The equation given is utilized for finite cloud correction when calculating external doses due to the airborne activity inside the control room.
4.3	The guidance provided in Regulatory Positions 4.1 and 4.2 should be used, as applicable, in re-assessing the radiological analyses identified in Regulatory Position 1.3.1, such as those in NUREG-0737 (Ref. 2). Design envelope source terms provided in NUREG-0737 should be updated for consistency with the AST. In general, radiation exposures to plant personnel identified in Regulatory Position 1.3.1 should be expressed in terms of TEDE. Integrated radiation exposure of plant equipment should be determined using the guidance of Appendix I of this guide.	Conforms	The Technical Support Center at PBAPS is in the Unit 1 Control Room. A review of the current TID-14844 based analysis indicates that it is unnecessary to reanalyze doses therein to assure accessibility. For other areas requiring plant personnel access, a qualitative assessment of the regulatory positions on source terms indicates that, with no new operator actions required, radiation exposures are bounded by those previously analyzed.
5.1.1	The evaluations required by 10 CFR 50.67 are re-analyses of the design basis safety analyses and evaluations required by 10 CFR 50.34; they are considered to be a significant input to the evaluations required by 10 CFR 50.92 or 10 CFR 50.59. These analyses should be prepared, reviewed, and maintained in accordance with quality assurance programs that comply with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50.	Conforms	These analyses were prepared as specified in the guidance.
5.1.2	Credit may be taken for accident mitigation features that are classified as safety-related, are required to be operable by technical	Conforms	The analyses take credit for SLC System operation. The SLC System is safety-related,

Table A: Conformance with Regulatory Guide (RG) 1.183 Main Sections			
RG Section	RG Position	PBAPS Analysis	Comments
	specifications, are powered by emergency power sources, and are either automatically actuated or, in limited cases, have actuation requirements explicitly addressed in emergency operating procedures. The single active component failure that results in the most limiting radiological consequences should be assumed. Assumptions regarding the occurrence and timing of a loss of offsite power should be selected with the objective of maximizing the postulated radiological consequences.		required to be operable by Technical Specifications, and supplied with emergency power . The SLC System is manually initiated from the main control room, as directed by the emergency operating procedures. There are four proceduralized injection methods for SLC with at least one alternate method for SLC injection that does not require personnel access into the secondary containment.
5.1.3	The numeric values that are chosen as inputs to the analyses required by 10 CFR 50.67 should be selected with the objective of determining a conservative postulated dose. In some instances, a particular parameter may be conservative in one portion of an analysis but be non-conservative in another portion of the same analysis.	Conforms	Conservative assumptions are used based on nominal values, as per prior plant analysis practice.
5.1.4	Licensees should ensure that analysis assumptions and methods are compatible with the AST and the TEDE criteria.	Conforms	Analysis assumptions and methods were made per this guidance.
5.3	Atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room that were approved by the staff during initial facility licensing or in subsequent licensing proceedings may be used in performing the radiological analyses identified by this guide. Methodologies that have been used for determining χ/Q values are documented in Regulatory Guides 1.3 and 1.4, Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," and the paper, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19". The NRC computer code PAVAN implements Regulatory Guide 1.145 and its use is acceptable to the NRC staff. The methodology of the NRC computer code ARCON96 is generally acceptable to the NRC staff for use in determining control room χ/Q values.	Conforms	New atmospheric dispersion values (χ/Q) for the EAB, the LPZ, and the control room were developed, using meteorological data for the years 1984-1988. ARCON96 and PAVAN were used with these data to determine control room and EAB/LPZ atmospheric dispersion values. Control room χ/Qs from releases from the Main Stack were developed in conformance with RG –1.194.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
1	Acceptable assumptions regarding core inventory and the release of radionuclides from the fuel are provided in Regulatory Position 3 of this guide.	Conforms	*Fission Product Inventory:* Core source terms are developed using ORIGEN-2.1 based methodology. *Release Fractions:* Release fractions are per Table 1 of RG 1.183, and are implemented by RADTRAD. *Timing of Release Phases:* Release Phases are per Table 4 of RG 1.183, and are implemented by RADTRAD. *Radionuclide Composition:* Radionuclide grouping is per Table 5 of RG 1.183, as implemented in RADTRAD. *Chemical Form:* Treatment of release chemical form is per RG 1.183, Section 3.5.
2	If the sump or suppression pool pH is controlled at values of 7 or greater, the chemical form of radioiodine released to the containment should be assumed to be 95% cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide. Iodine species, including those from iodine re-evolution, for sump or suppression pool pH values less than 7 will be evaluated on a case-by-case basis. Evaluations of pH should consider the effect of acids and bases created during the LOCA event, e.g., radiolysis products. With the exception of elemental and organic iodine and noble gases, fission products should be assumed to be in particulate form.	Conforms	The stated distributions of iodine chemical forms are used. The post-LOCA suppression pool pH has been evaluated, including consideration of the effects of acids and bases created during the LOCA event, the effects of key fission product releases, and the impact of SLC injection. Suppression pool pH remains above 7 for at least 30 days.
3.1	The radioactivity released from the fuel should be assumed to mix instantaneously and homogeneously throughout the free air volume of the primary containment in PWRs or the drywell in BWRs as it is released. This distribution should be adjusted if there are internal compartments that have limited ventilation exchange. The suppression pool free air volume may be included provided there is a mechanism to	Conforms	The radioactivity release from the fuel is assumed to homogeneously mix only in the drywell free volume during the first 2 hours after the assumed LOCA, and in the combined drywell free air volume and

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	ensure mixing between the drywell to the wetwell. The release into the containment or drywell should be assumed to terminate at the end of the early in-vessel phase.		suppression chamber air space for the 2 to 720 hour period. Mixing is caused by steam flashing and flow from the drywell through the suppression pool to the suppression chamber air space, after core reflood.
3.2	Reduction in airborne radioactivity in the containment by natural deposition within the containment may be credited. Acceptable models for removal of iodine and aerosols are described in Chapter 6.5.2, "Containment Spray as a Fission Product Cleanup System," of the Standard Review Plan (SRP), NUREG-0800 (Ref. A-1) and in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments" (Ref. A-2). The latter model is incorporated into the analysis code RADTRAD (Ref. A-3).	Conforms	Credit is taken for natural deposition per the methodology of NUREG/CR-6189, as implemented in RADTRAD. No deterministically assumed initial plateout is credited.
3.3	Reduction in airborne radioactivity in the containment by containment spray systems that have been designed and are maintained in accordance with Chapter 6.5.2 of the SRP (Ref. A-1) may be credited. Acceptable models for the removal of iodine and aerosols are described in Chapter 6.5.2 of the SRP and NUREG/CR-5966, "A Simplified Model of Aerosol Removal by Containment Sprays"1 (Ref. A-4). This simplified model is incorporated into the analysis code RADTRAD (Refs. A-1 to A-3). The evaluation of the containment sprays should address areas within the primary containment that are not covered by the spray drops. The mixing rate attributed to natural convection between sprayed and unsprayed regions of the containment building, provided that adequate flow exists between these regions, is assumed to be two turnovers of the unsprayed regions per hour, unless other rates are justified. The containment building atmosphere may be considered a single, well-mixed volume if the spray covers at least 90% of the volume and if adequate mixing of unsprayed compartments can be shown. The SRP sets forth a maximum decontamination factor (DF) for	Not Applicable	While containment sprays are a design feature that is available at PBAPS, no credit is taken for aerosol removal by them in the LOCA AST reanalysis.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	elemental iodine based on the maximum iodine activity in the primary containment atmosphere when the sprays actuate, divided by the activity of iodine remaining at some time after decontamination. The SRP also states that the particulate iodine removal rate should be reduced by a factor of 10 when a DF of 50 is reached. The reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass. There is no specified maximum DF for aerosol removal by sprays. The maximum activity to be used in determining the DF is defined as the iodine activity in the columns labeled "Total" in Tables 1 and 2 of this guide multiplied by 0.05 for elemental iodine and by 0.95 for particulate iodine (i.e., aerosol treated as particulate in SRP methodology).		
3.4	Reduction in airborne radioactivity in the containment by in-containment recirculation filter systems may be credited if these systems meet the guidance of Regulatory Guide 1.52 and Generic Letter 99-02 (Refs. A-5 and A-6). The filter media loading caused by the increased aerosol release associated with the revised source term should be addressed.	Not Applicable	No in-containment recirculation filter systems exist at PBAPS.
3.5	Reduction in airborne radioactivity in the containment by suppression pool scrubbing in BWRs should generally not be credited. However, the staff may consider such reduction on an individual case basis. The evaluation should consider the relative timing of the blowdown and the fission product release from the fuel, the force driving the release through the pool, and the potential for any bypass of the suppression pool (Ref. 7). Analyses should consider iodine re-evolution if the suppression pool liquid pH is not maintained greater than 7.	Conforms	No credit is taken for suppression pool scrubbing in the LOCA AST reanalysis. Analyses have been performed that determined that the suppression pool liquid pH is maintained greater than 7, and that, therefore, iodine re-evolution is not expected.
3.6	Reduction in airborne radioactivity in the containment by retention in ice condensers, or other engineering safety features not addressed above, should be evaluated on an individual case basis. See Section 6.5.4 of the SRP (Ref. A-1).	Not Applicable	PBAPS does not have ice condensers. No other removal mechanisms are credited other than natural deposition.
3.7	The primary containment (i.e., drywell for Mark I and II containment designs) should be assumed to leak at the peak pressure technical specification leak rate for the first 24 hours. For PWRs, the leak rate may be reduced after the first 24 hours to 50% of the technical	Conforms	Primary containment leakage is assumed to be at the 0.7% of containment mass per day for 24 hours, 0.392% from 24 to 38 hours,

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	specification leak rate. For BWRs, leakage may be reduced after the first 24 hours, if supported by plant configuration and analyses, to a value not less than 50% of the technical specification leak rate. Leakage from subatmospheric containments is assumed to terminate when the containment is brought to and maintained at a subatmospheric condition as defined by technical specifications. For BWRs with Mark III containments, the leakage from the drywell into the primary containment should be based on the steaming rate of the heated reactor core, with no credit for core debris relocation. This leakage should be assumed during the two-hour period between the initial blowdown and termination of the fuel radioactivity release (gap and early in-vessel release phases). After two hours, the radioactivity is assumed to be uniformly distributed throughout the drywell and the primary containment.		and 0.35% per day from 38 to 720 hours. This is based on the results of the leak characteristic methodology evaluation performed (turbulent flow). The Darcy's Formula evaluation methodology is considered the most conservative approach for the evaluation of the primary containment leak rate. The large break LOCA was found to be the bounding containment pressurization event. Even if a LOCA were to occur during purging, isolation valve closure would occur within a small fraction of the time before start of the gap release. Dose due to this purge would be negligible as compared to other dose contributors. PBAPS uses a Mark I containment.
3.8	If the primary containment is routinely purged during power operations, releases via the purge system prior to containment isolation should be analyzed and the resulting doses summed with the postulated doses from other release paths. The purge release evaluation should assume that 100% of the radionuclide inventory in the reactor coolant system liquid is released to the containment at the initiation of the LOCA. This inventory should be based on the technical specification reactor coolant system equilibrium activity. Iodine spikes need not be considered. If the purge system is not isolated before the onset of the gap release phase, the release fractions associated with the gap release and early in-vessel phases should be considered as applicable.	Conforms	The PBAPS primary containment is not routinely purged during power operation. Purging is limited to inerting, de-inerting and occasional short pressure control activities.
4.1	Leakage from the primary containment should be considered to be collected, processed by engineered safety feature (ESF) filters, if any, and released to the environment via the secondary containment exhaust	Conforms	Secondary Containment elevated release via the Main Stack credit is taken at 15 minutes after the start of

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	system during periods in which the secondary containment has a negative pressure as defined in technical specifications. Credit for an elevated release should be assumed only if the point of physical release is more than two and one-half times the height of any adjacent structure.		gap release. Gap release begins at ~ 2 minutes after LOCA initiation. For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity RB/TB vent stack release assumptions, yielding ground level release equivalent dispersion factors.
4.2	Leakage from the primary containment is assumed to be released directly to the environment as a ground-level release during any period in which the secondary containment does not have a negative pressure as defined in technical specifications.	Conforms	For EAB and LPZ doses, ground level releases are assumed. For Control Room doses, releases are based on zero-velocity RB/TB vent stack release assumptions.
4.3	The effect of high wind speeds on the ability of the secondary containment to maintain a negative pressure should be evaluated on an individual case basis. The wind speed to be assumed is the 1-hour average value that is exceeded only 5% of the total number of hours in the data set. Ambient temperatures used in these assessments should be the 1-hour average value that is exceeded only 5% or 95% of the total numbers of hours in the data set, whichever is conservative for the intended use (e.g., if high temperatures are limiting, use those exceeded only 5%).	Conforms	The wind speed exceeded only 5% of the time at PBAPS in the secondary containment vicinity is approximately 11 mph. It has been determined that a 23 mph wind speed would be required before the secondary containment pressures would be positive relative to outside air pressures at the downwind side of the reactor enclosure.
4.4	Credit for dilution in the secondary containment may be allowed when adequate means to cause mixing can be demonstrated. Otherwise, the leakage from the primary containment should be assumed to be transported directly to exhaust systems without mixing. Credit for mixing, if found to be appropriate, should generally be limited to 50%. This evaluation should consider the magnitude of the containment	Conforms	No credit is taken for dilution/mixing in secondary containment. An artificially small secondary containment volume is assumed in the RADTRAD analysis in conjunction with a large SGTS flow

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	leakage in relation to contiguous building volume or exhaust rate, the location of exhaust plenums relative to projected release locations, the recirculation ventilation systems, and internal walls and floors that impede stream flow between the release and the exhaust.		rate to ensure mixing is not an issue.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
4.5	Primary containment leakage that bypasses the secondary containment should be evaluated at the bypass leak rate incorporated in the technical specifications. If the bypass leakage is through water, e.g., via a filled piping run that is maintained full, credit for retention of iodine and aerosols may be considered on a case-by-case basis. Similarly, deposition of aerosol radioactivity in gas-filled lines may be considered on a case-by-case basis.	Conforms	No primary containment leakage, with the exception of MSIV leakage, has been identified which bypasses the secondary containment. Only the MSIV pathway leak rates are incorporated into the Technical Specifications. The AST analysis is based on an MSIV leakage limit of 150 scfh total leakage with not more that 75 scfh per line when tested at ≥ 25 psig. The dose consequences for releases through this pathway (with piping deposition credit) are separately calculated. Therefore, MSIV leakage can continue to be excluded from Type B and C leakage total evaluated against the revised L_a of 0.7% per day. Piping deposition credit is determined using the AEB 98-03 well-mixed method. Delay in transit through these piping system is not credited. The credited piping is that which has previously been seismically qualified and is from the reactor vessel to the Turbine Stop Valves. However, consistent with an assumption of a LOCA in a main steam line inside containment, the most beneficial line for deposition is assumed to have the break and also have its inboard MSIV failed. In consideration of possible turbulence

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
			in containment in the failed MSIV vicinity, the first two pipe diameters of penetration piping are not credited for this line. The balance of penetration piping is treated as inboard piping.
4.6	Reduction in the amount of radioactive material released from the secondary containment because of ESF filter systems may be taken into account provided that these systems meet the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	SGTS HEPA and charcoal adsorber filters are not credited in the evaluation of analyzed accidents onsite and offsite dose consequences.
5.1	With the exception of noble gases, all the fission products released from the fuel to the containment (as defined in Tables 1 and 2 of this guide) should be assumed to instantaneously and homogeneously mix in the primary containment sump water (in PWRs) or suppression pool (in BWRs) at the time of release from the core. In lieu of this deterministic approach, suitably conservative mechanistic models for the transport of airborne activity in containment to the sump water may be used. Note that many of the parameters that make spray and deposition models conservative with regard to containment airborne leakage are non-conservative with regard to the buildup of sump activity.	Conforms	With the exception of noble gases, all the fission products released from the fuel to the containment are assumed to instantaneously and homogeneously mix only in the drywell during the first 2 hours after the assumed LOCA, and in the combined drywell and suppression chamber free volume for the 2 to 720 hour period.
5.2	The leakage should be taken as two times the sum of the simultaneous leakage from all components in the ESF recirculation systems above which the technical specifications, or licensee commitments to item III.D.1.1 of NUREG-0737 (Ref. A-8), would require declaring such systems inoperable. The leakage should be assumed to start at the earliest time the recirculation flow occurs in these systems and end at the latest time the releases from these systems are terminated. Consideration should also be given to design leakage through valves isolating ESF recirculation systems from tanks vented to atmosphere, e.g., emergency core cooling system (ECCS) pump miniflow return to the refueling water storage tank.	Conforms	The 5 gpm leak rate is assumed to be two times the sum of the simultaneous leakage from all ECCS components as discussed in the dose calculations. ECCS leakage is minimized at PBAPS through implementation of the Program committed to in T.S. 5.5.2 "Primary Coolant Sources Outside Containment". Since certain ECCS systems take suction immediately from the

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
			suppression pool, this leak path is assumed to start at time 0. Leakage to atmospheric tanks is credible only for lines connecting from ECCS pump discharges to such a tank, because of relative elevations. The sole leakage paths to a tank vented to atmosphere meeting this condition are the High Pressure Coolant Injection / Reactor Core Isolation Cooling test lines that discharge to the Condensate Storage Tank (CST). These lines are isolated by two normally closed valves. Since the CST contents are demineralized water, ECCS leakage would quickly turn the water basic. Therefore, minimal elemental iodine is expected, and as a result, negligible iodine volatilization.
5.3	With the exception of iodine, all radioactive materials in the recirculating liquid should be assumed to be retained in the liquid phase.	Conforms	With the exception of iodine, all radioactive materials in ECCS liquids are assumed to be retained in the liquid phase.
5.4	If the temperature of the leakage exceeds 212°F, the fraction of total iodine in the liquid that becomes airborne should be assumed equal to the fraction of the leakage that flashes to vapor. This flash fraction, FF, should be determined using a constant enthalpy, h, process, based on the maximum time-dependent temperature of the sump water circulating outside the containment: $FF = \frac{h_{f1} - h_{f2}}{h_{fg}}$ *Where:* h_{f1} is the enthalpy of liquid at system design temperature and	Not Applicable	The temperature of the leakage does not exceed 212°F.

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	pressure; h_{f2} is the enthalpy of liquid at saturation conditions (14.7 psia, 212°F); and h_{fg} is the heat of vaporization at 212°F.		
5.5	If the temperature of the leakage is less than 212°F or the calculated flash fraction is less than 10%, the amount of iodine that becomes airborne should be assumed to be 10% of the total iodine activity in the leaked fluid, unless a smaller amount can be justified based on the actual sump pH history and area ventilation rates.	Conforms	An airborne release fraction of 1.41% is used. Suppression Pool water pH is maintained above 7 for the entire 30 days of the accident dose assessment period. Under these conditions virtually none of the iodine will be in elemental form, and organic iodine formation will be inhibited. Because of the subcooled condition no flashing is expected. Nevertheless, this value, derived based on ORNL-TM-2412 methodology for iodine partition factor determination, is used.
5.6	The radioiodine that is postulated to be available for release to the environment is assumed to be 97% elemental and 3% organic. Reduction in release activity by dilution or holdup within buildings, or by ESF ventilation filtration systems, may be credited where applicable. Filter systems used in these applications should be evaluated against the guidance of Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	The credited Control Room intake charcoal and HEPA filters meet the requirements of RG 1.52 and Generic Letter 99-02. These are credited at 90% efficiency for elemental and organic iodines. Aerosol removal efficiencies are assumed to be 99% based on the HEPA/charcoal combination.
6.1	For the purpose of this analysis, the activity available for release via MSIV leakage should be assumed to be that activity determined to be in the drywell for evaluating containment leakage (see Regulatory Position 3). No credit should be assumed for activity reduction by the steam separators or by iodine partitioning in the reactor vessel.	Conforms	The radioactivity release from the fuel is assumed to homogeneously mix only in the drywell free volume during the first 2 hours after the assumed LOCA, and in the combined drywell free air volume and suppression chamber air space for the 2 to 720 hour period. Mixing is

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
			caused by steam flashing and flow from the drywell through the suppression pool to the suppression chamber air space, after core reflood.
6.2	All the MSIVs should be assumed to leak at the maximum leak rate above which the technical specifications would require declaring the MSIVs inoperable. The leakage should be assumed to continue for the duration of the accident. Postulated leakage may be reduced after the first 24 hours, if supported by site-specific analyses, to a value not less than 50% of the maximum leak rate.	Conforms	MSIV leakage assumed in this accident analysis is 150 scfh for all steam lines and 75scfh for any one line when tested at ≥ 25 psig. Reduction in leakage rates after 24 hours are, as previously discussed, based on calculated post-accident containment pressures. No credit is taken for leakage rate reductions below 50% of the MSIV leakage limit.
6.3	Reduction of the amount of released radioactivity by deposition and plateout on steam system piping upstream of the outboard MSIVs may be credited, but the amount of reduction in concentration allowed will be evaluated on an individual case basis. Generally, the model should be based on the assumption of well-mixed volumes, but other models such as slug flow may be used if justified.	Conforms	Modeling is per AEB 98-03 well-mixed approach, with no transport delay credit.
6.4	In the absence of collection and treatment of releases by ESFs such as the MSIV leakage control system, or as described in paragraph 6.5 below, the MSIV leakage should be assumed to be released to the environment as an unprocessed, ground- level release. Holdup and dilution in the turbine building should not be assumed.	Conforms	Releases are assumed to be from the RB/TB vent stacks, without credit for holdup or dilution in the condenser or turbine building. The zero velocity RB/TB vent stacks release assumption is effectively a ground level release assumption.
6.5	A reduction in MSIV releases that is due to holdup and deposition in main steam piping downstream of the MSIVs and in the main condenser, including the treatment of air ejector effluent by offgas systems, may be credited if the components and piping systems used in the release path are capable of performing their safety function during	Conforms	Non-faulted main steam piping that is capable of performing its safety function during and following an SSE is credited. No credit is taken for holdup and deposition in piping

Table B: Conformance with RG 1.183 Appendix A (Loss-of-Coolant Accident)			
RG Section	RG Position	PBAPS Analysis	Comments
	and following a safe shutdown earthquake (SSE). The amount of reduction allowed will be evaluated on an individual case basis. References A-9 and A-10 provide guidance on acceptable models.		downstream of the qualified main steam piping, or in the condenser. The modeling is per the AEB 98-03 well-mixed approach.
7.0	The radiological consequences from post-LOCA primary containment purging as a combustible gas or pressure control measure should be analyzed. If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated. If the primary containment purging is required within 30 days of the LOCA, the results of this analysis should be combined with consequences postulated for other fission product release paths to determine the total calculated radiological consequences from the LOCA. Reduction in the amount of radioactive material released via ESF filter systems may be taken into account provided that these systems meet the guidance in Regulatory Guide 1.52 (Ref. A-5) and Generic Letter 99-02 (Ref. A-6).	Conforms	Containment purging as a combustible gas or pressure control measure is not required nor credited in any design basis analysis for 30 days following a design basis LOCA at PBAPS.

ATTACHMENT 7

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
"PBAPS Alternative Source Term Implementation"

Post-Accident Vital Area Access Considerations Table Revision

Table 2: Post-Accident Vital Area Access Considerations (not including occupancy for the Control Room and Technical Support Center)

Activity/Access Route	Current Design Bases	Bases for Post-AST Accessibility
Projected Doses to Individuals Accessing Vital Areas Requiring Continuous Occupancy such as the control room and TSC	Based on exterior dose rates and travel times, typically from Guard House to TSC or Control Room. Bounding dose is 0.295 whole body. Inhalation doses were not evaluated for this pathway.	Doses have been re-analyzed with AST source terms based on worst 4-hour release rates and include cloud doses and unshielded refuel floor shine and are between 0.75 and 0.82 rem. SCBA use is assumed (i.e., no inhalation dose).
Projected Doses to Individual for Necessary Access to and Infrequent Occupancy of Vital Areas within Turbine Hall/Radwaste Building Complex (Chem Lab/Counting Room, PASS, Radwaste Control Room, and Cable Spread Room)	Based on Travel time of 10 minutes and 8 hours of continuous occupancy the currently analyzed whole body dose is 1.675 rem.	A review of identified functions indicated that 8 hours is an excessive allowance. This occupancy assumption has been reduced to 1 hour except for 2 hours in the shielded Chem Lab/Counting Room. Dose assessments include unshielded cloud shine (no geometry factor credit) with resulting doses at a maximum of 0.295 rem for access and 1.77 rem maximum for occupancy.
Projected Total Whole Body Dose to Individuals for Necessary Access to and Occupancy of Diesel-Generator Building	Currently analyzed dose is 0.433 rem whole body, with doses dominated by 16 minute travel time due to DG Building Shielding.	RADTRAD analyses, using control room X/Qs to conservatively simulate the site in general, yield a peak TEDE dose rate of 1.8 rem/hr. For the subject 13 minute travel time the resulting dose would be 0.576 rem including 30 minute occupancy inside the shielded D/G building.
Projected Total Whole Body Dose to Individuals for Necessary Access to and Occupancy of the CAD Building outside Reactor Building.	See to the right	Access to the CAD building is no longer required since the former Containment Atmosphere Dilution System post LOCA combustible gas control function is no longer required.

Table 2: Post-Accident Vital Area Access Considerations (not including occupancy for the Control Room and Technical Support Center)

Activity/Access Route	Current Design Bases	Bases for Post-AST Accessibility
Projected Total Whole Body Dose to Individuals for Necessary Access to and Occupancy of the Cartridge Exchange at the RAD Effluent Stack Monitor.	Currently analyzed dose is 4.981 rem whole body, with doses dominated by shine from refuel floor airborne activity and with a significant contribution from equipment/piping shine.	The location of this monitor has previously been moved to the Turbine Building El. 195 from the original location on the Reactor Building El. 234. Additionally, a more direct pathway starting from the control room, to the monitor, and then to the Sample Analysis Chem Lab on Turbine Building El. 116 has been selected, significantly reducing operator dose. The calculated dose is 1.84 rem.

Table 2: Post-Accident Vital Area Access Considerations (not including occupancy for the Control Room and Technical Support Center)

Activity/Access Route	Current Design Bases	Bases for Post-AST Accessibility
Projected Total Whole Body Dose to Individuals for Necessary Access to and Occupancy of the Makeup Water to Spent Fuel Pools at the RAD Effluent Stack Monitor.	Currently analyzed dose is 4.952 rem whole body, with doses dominated by shine from refuel floor airborne activity and with a significant contribution from equipment/piping shine.	The original analysis was performed based on provision of spent fuel pool makeup on conditions associated with a LOCA at 2 hours after the event. The need for the action at that time was not addressed, nor was refuel floor accessibility. This issue has been reassessed and accessibility is no longer considered necessary based on the following: • If spent fuel cooling is lost due to loss of offsite power, immediately after a refueling outage, at least 24 hours would be required before the onset of spent fuel pool boiling. • On the order of an additional 72 hours would be required to lose 10 feet of water coverage over the stored spent fuel. This would leave on the order of 13 feet of water coverage for shielding over the fuel. • Regulatory Guide 1.155 supporting documentation such as NUREG-1109 and NSAC-103 indicated that the median loss of offsite power duration is 0.5 hours, with restoration within 3 hours 90% of the time, and the maximum observed duration of 9 hours. More recent, and especially severe, loss of offsite power durations such as the wide-spread August 14, 2003 grid disturbance had offsite power restoration (Fermi) within 21 hours. • Ample time is thus available for off-site power restoration for the important but non-safety related function of spent fuel pool level control and cooling.

ATTACHMENT 8

PEACH BOTTOM ATOMIC POWER STATION
UNITS 2 AND 3

Docket Nos. 50-277
50-278

License Nos. DPR-44
DPR-56

Supplement to License Amendment Request for
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LOCA Radiological Consequences Analysis Revision

Table 9: LOCA Radiological Consequence Analysis			
Location	**Duration**	**TEDE (rem)**	**Regulatory Limit TEDE (rem)**
Control Room	30 days	4.59*	5
EAB	Maximum, 2 hours	8.17	25
LPZ	30 days	4.99	25

* The doses here include the direct shine and inhalation doses from radioactivity drawn into the control room as well as the dose from external sources. Dose is based on an assumed MSIV total leakage of 150 scfh, which contributes 3.922 rem to the total. The other contributions are 0.606 from Primary Containment leakage, 0.009 from ECCS leakage in Secondary Containment, and 0.050 rem from gamma shine.