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10CFR51

January 30, 2002

U.S. Nuclear Regulatory Commission
ATTN: L. Wheeler – Mail Stop O11-F1
Washington, DC 20555

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

Subject: Response to Request for Additional Information Related to Severe Accident Mitigation Alternatives

Reference: Letter from L. L. Wheeler (USNRC) to M. P. Gallagher (Exelon), dated December 20, 2001

Dear Sir/Madam:

Exelon Generation Company, LLC (Exelon) hereby submits the enclosed responses to the request for additional information transmitted in the reference letter. For your convenience, attachment 1 restates the questions from the reference letter and provides our responses.

If you have any questions or require additional information, please do not hesitate to call.

Very truly yours,

D. G. Helker /for

Michael P. Gallagher
Director, Licensing and Regulatory Affairs
Mid-Atlantic Regional Operating Group

Enclosures: Affidavit, Attachment 1

cc: H. J. Miller, Administrator, Region I, USNRC (w/o enc)
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS (w/o enc)

[illegible]
add: Duke Wheeler
to encls

Affidavit of Joseph J. Hagan

I, Joseph J. Hagan, Senior Vice President, do hereby affirm and state:

1. I am authorized to execute this affidavit on behalf of Exelon Generation Company, LLC ("EGC").

2. EGC is providing this information is support of its Application for License Renewal for the Peach Bottom Atomic Power Station Units 2 and 3 (NRC Facility Operating License Nos. DPR-44 and DPR-56; Docket Nos. 50-277 and 50-278.)

3. I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

J J Hagan

Joseph J. Hagan
Senior Vice President

Commonwealth of Pennsylvania
County of Chester

Subscribed and sworn to before me, a Notary Public, in and for the County and Commonwealth above named, this 23rd day of January, 2002.

Vivia V. Gallimore

Notary Public

My Commission Expires:

10-6-03

Notarial Seal
Vivia V. Gallimore, Notary Public
Kennett Square Boro, Chester County
My Commission Expires Oct. 6, 2003
Member, Pennsylvania Association of Notaries

ATTACHMENT 1

1. ***Please provide the following information related to the Peach Bottom Unit 2 PSA (PB99, Rev. 1) that forms the basis for the Severe Accident Mitigation Alternative (SAMA) analysis:***

   a. ***A description of the major differences from the level 1 and 2 Individual Plant Examination (IPE) previously reviewed by the staff, and the respective impacts of these changes on core damage frequency (CDF) and release frequency. Specifically address the reasons for a factor of 2 reduction in the internal events CDF in the SAMA submittal (total CDF of about $2.3x10^{-6}$ per reactor-year) as compared with the IPE (total CDF of about $5.5x10^{-6}$ per reactor-year),***

   b. ***A description of the internal and external peer review process used for the updated risk study,***

   c. ***A breakdown in the contributions of various accident types to the CDF. Based on the information provided for the various plant damage states, the following breakdown is inferred: LOCAs, $1.0x10^{-7}$; transients, $1.3x10^{-7}$; station blackout, $1.02x10^{-6}$; ATWS, $1.10x10^{-6}$. Please confirm and/or provide the correct breakdown, and***

   d. ***The core damage frequency and large release frequency for Peach Bottom Unit 3, and the major reasons why these values are lower than for Unit 2.***

<u>Response to RAI 1a</u>

The total PBAPS Unit 2 Level 1 CDF used in the SAMA submittal is 4.53E-06 per reactor year. The frequency associated with the plant damage state (PDS) releases is 2.36E-06 per reactor year. The difference between the Level 1 CDF and the Level 2 endstate frequency represents those core damage sequences that lead to negligible or no release from the primary containment.

The difference between the current PBAPS Unit 2 CDF and the IPE CDF is approximately 20%. Two major model updates were performed, in 1997 and the other in 1999. The more significant changes to the PSA models and subsequent changes in results are summarized as follows:

- Better plant operating experience was reflected in the overall frequency of initiating events

- Subsumed initiating events such as the loss of instrument air, service water, etc. were modeled as separate initiating events

- Detailed modeling of operator actions directed by procedures during a loss of offsite power (LOOP)

- Common cause re-evaluation using the new Idaho National Engineering and Environmental Laboratory (INEEL) database

- Incorporated Improved Tech Spec changes

- Accounted for the Conowingo tie-line in the LOOP models

- Re-evaluated LOOP recoveries and associated timing in the event sequence modeling

- Added common cause failure terms for HPCI/RCIC, DC battery pairs, and other miscellaneous systems

The incorporation of lower initiating frequencies, additional LOOP recovery capabilities such as the Conowingo tie-line, and the INEEL common cause database resulted in reducing the total CDF from that reported in the IPE. The modeling of additional initiating events, detailed operator actions for LOOP, and common cause terms for HPCI/RCIC and DC batteries had the effect of increasing the total CDF. The incorporation of all modeling and data changes resulted in an overall reduction in total CDF.

Response to RAI 1b

Individual work packages or model development tasks were assigned to either utility or vendor personnel. Another individual independently peer reviewed the work. Integrated evaluations of the updated models were reviewed and calibrated using multiple individuals. Separate Unit models were also quantified and compared to evaluate known asymmetries between the Units.

The Level 1 and Level 2 PBAPS PSA models were reviewed by a team of 6 outside reviewers in 1996 and again in 1998 using the BWROG PSA Peer Review Certification Implementation Guidelines. The purpose of the certification process was to establish confidence in the technical quality of the PSA for a spectrum of applications. The PBAPS PSA certification teams consisted of 3 BWR utility and 3 PSA vendor personnel. The combined PSA expertise of each review team was approximately 100 years. Fact and Observation sheets documented the certification team's insights and potential level of significance. These insights do not materially affect the overall risk calculation and therefore the cost-benefit assessment. The recent PSA model updates, however, incorporated changes to address the significant certification issues.

Response to RAI 1c

The contribution of the various accident types to the total PBAPS Unit 2 CDF is provided in the following table.

Contributor to CDF	CDF	% Contribution
LOOP	2.07E-06	45.8
Transients	1.25E-06	27.6
SBO	4.68E-07	10.4
ATWS	4.33E-07	9.6
LOCA	1.94E-07	4.3
Flood	5.96E-08	1.3
Other	4.77E-08	1.1
TOTAL	**4.53E-06**	**100**

Response to RAI 1d

The comparison of PBAPS Unit 2 and Unit 3 CDF and large early release frequency (LERF) and reasons for the differences are as follows:

Unit 2		Unit 3	
CDF	**LERF**	**CDF**	**LERF**
4.53E-06	6.17E-08	4.18E-06	5.90E-08

The 3.5E-07 (8%) difference in CDF between the Units is attributed mostly to Loss of Offsite Power (LOOP) sequences involving the loss of 2 or 3 shared diesel generators. Asymmetry in emergency electric power distribution between the Units and the diesel loading capability (one RHR pump per diesel generator) concurrent with the common LOOP initiator result in different diesel failure combinations having different CDF impacts. The difference in LERF is proportional to the difference in CDF.

2. ***Based on the discussion in Section 4.20.2.2, it appears that the site specific economic and agricultural characteristics (e.g., land values) used in the SAMA analysis were assumed to be a factor of four greater than used in NUREG/CR-4551. Please confirm this.***

Response to RAI 2

Section 4.20.2.2 indicates "It is assumed that the relative distribution of these factors has remained constant and that the overall growth in "economy" and "agriculture" is represented by the growth in population. This growth is reflected by means of scaling the Offsite Economic Cost Risk by the increase in population".

The population estimate for the area surrounding the site used in the NUREG/CR-4551 analysis was originally based on 1980 census information. This SAMA evaluation requires an estimate of the 50-mile population at the end of the license extension in 2034. For the purposes of the analysis, the 2034 population was estimated using data from the 1980 and 1990 census and a simple, linear growth approximation for the population density in the surrounding area. The calculated increase in population within 50 miles was determined to be 3.99 (a factor of 4) times greater than that used in NUREG/CR-4551.

3. ***In assessing the costs associated with core damage events at Peach Bottom, neither the impact of uncertainties nor the contribution of external initiators (e.g., seismic, fires, etc.) have been considered. In this regard, please provide the following:***

   a. ***An assessment of the uncertainties associated with the calculated core damage frequency (e.g., the mean and median CDF estimates and the 5^{th} and 95^{th} percentile values of the uncertainty distribution), and the impact on SAMA identification and screening results if risk reduction estimates were based on the upper end of the distribution rather than the mean value,***

   b. ***An assessment of the impact of including the risk from external events in the SAMA identification and screening process. It is recognized that the methods used for the Peach Bottom IPEEE do not provide numerical estimates of the CDF contributions from seismic and fire initiators; however, quantitative estimates for CDF and risk for external events at Peach Bottom are available in NUREG/CR-4551, and can be used to account for the impact of external events in the SAMA analysis. This impact can be substantial since the risk associated with external events in NUREG/CR-4551 (e.g., 57 person-rem per reactor year for fires) is much greater than the total risk estimate used in the SAMA analysis (14.7 person-rem per reactor year),***

   c. ***Explanation of whether low cost SAMAs screened out in the***

analysis (e.g., Phase II SAMA numbers 1, 13, 21 in Table G.4-2) would become cost beneficial if the screening were conservatively based on the upper bound of the benefit (i.e., considering the uncertainties in risk estimates and the contribution from external events), and

d. Clarification whether the implementation costs and net values reported in Section 4.20.6 and Table G.4-2 reflect the value for Unit 2 or the combined values for both units, and confirmation that this is consistent with the baseline costs of a severe accident ($2.04 million) as used in the screening process (which is based on two units).

Response to RAI 3a

There are uncertainties in all of the inputs to the cost benefit analysis, for example:

- The cost of modifications can increase substantially as additional levels of detail are added
- The consequences of an accident may be less than modeled due to effective mitigation
- The effectiveness of a modification may be overestimated in the credit it receives

In recognition of the potential for uncertainties in all inputs, the NRC Regulatory Analysis Guidelines[1] and NSAC Guidance on Cost-Benefit Analysis[2] both clearly state that the evaluation is to be performed with best estimate values to avoid distorting or biasing the results.

Example guidance is found as follows:

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(1) *Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission*, NUREG/BR-0058, Revision 2, Nuclear Regulatory Commission, 1992.

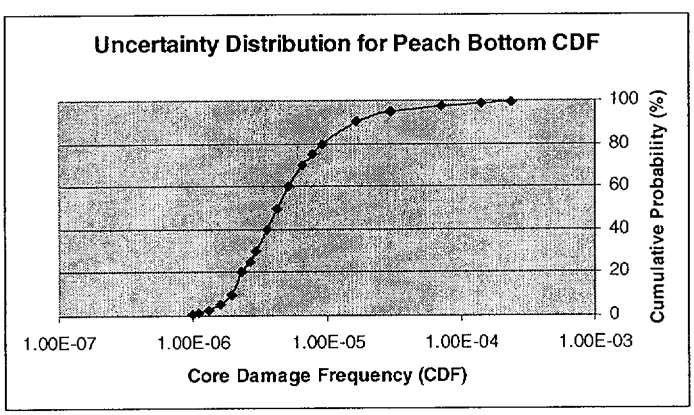
(2) W. Reuland, H. Wyckoff, *Questionable Techniques Used in Cost-Benefit Analyses of Nuclear Safety Enhancements*, NSAC-143, November 1989.

- Value and impact estimates are to be incremental best estimates relative to the baseline case, which is normally the no-action alternative. When possible, best estimates should be made in terms of the "mean" or "expected value." However, other acceptable estimates could include median and point estimates, depending upon the level of detail available from the data sources employed in the value/impact analysis. (p. 4-9)[1]
- Relevant value attributes should be identified and assessed for each alternative. These assessments should reflect best estimates, preferably mean values. (p. 4-12)[1]
- Suffice it to say, the analyst should strive to develop and use best estimate core melt and person-rem probabilities. Core melt frequencies and person-rem exposure based on conservative rather than best estimate probabilities are distorted. (p. 2-11)[2]

Nevertheless, the recognition of uncertainties in the cost-benefit evaluation is important. The requested use of a 95% upper bound on one of the inputs, i.e., the CDF, is one example of a sensitivity that can provide additional perspective on the analysis for use by decision-makers. This perspective must, in turn, be balanced by the insight to be gained if the CDF is really at the 5% lower bound or the costs are at their 95% upper bound.

A Monte-Carlo propagation of data uncertainty was performed for the Peach Bottom Unit 2 PSA model using the WinNUPRA™ software. The results of that analysis are summarized below.

Confidence	CDF
5th	1.57E-06
25th	2.63E-06
50th	4.24E-06
75th	7.76E-06
95th	3.00E-05



The 95th percentile associated with this distribution is 3.00E-05/yr. The SAMA baseline CDF of 4.5E-06/yr corresponds to approximately the 55th percentile of this CDF distribution. There are many factors to consider, however, when looking at the benefits of the SAMA candidates. Plant specific implementation of SAMA candidates may be complicated by space limitations, outage costs, regulatory requirements, and other considerations. These factors tend to result in underestimation of the costs. Additionally, the specific PSA analyses that were performed in addressing specific SAMA candidates were done optimistically. That is, the potential cost-benefit was derived from a case that maximized the CDF reduction that would result from implementation of the SAMA. Both of these factors would, in effect, offset the uncertainties associated with the CDF estimates.

In any event, the Phase I SAMA screening results were reviewed to see if any of the conclusions would change if the 95th percentile CDF value were referenced instead of the mean value. It was found that none of the initial Phase I dispositions were dependent upon the actual CDF value used. However, the Phase II dispositions were found to be dependent upon the CDF value used since this would impact the baseline cost as well as the potential cost reduction.

With a CDF value of 3.00E-05 instead of 4.53E-06, and assuming that the Level 2 results would propagate at the same ratio (i.e., 3.00E-05/4.53E-06=6.62), the baseline cost of a severe accident represented by this sensitivity becomes nearly \$6.8M per unit instead of the previously calculated \$1M per unit. Where available, the cost-benefit was re-performed considering the change in the cost basis from \$2M to \$13.5M per site. In cases where a detailed cost-benefit analysis had not been previously employed, either a new analysis was performed or a first-order approximation of the benefit was obtained by considering the contribution to CDF that could be averted and comparing the associated cost-benefit from that to the revised baseline cost of \$13.5M. Table 3-1 summarizes the results of this sensitivity study of the Phase II SAMA disposition using this process. Note that the first six columns of Table 3-1 are reproduced here from Table G.4-2 of the submittal, and that the last column of Table 3-1 has been added to show the results of this sensitivity study.

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
1	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	SAMA would reduce the frequency of the loss of component cooling water and service water.	$50K	Assume $50K for site procedure change	Detailed cost-benefit analysis performed. Net value of -$41,591 indicates that the SAMA is not beneficial. Refer to section G.5.1.	*Revised cost-benefit analysis using the 95th percentile CDF was performed. Net value of -$11.1K indicates that the SAMA is still not beneficial.*
2	Improved ability to cool the residual heat removal heat exchangers.	SAMA would reduce the probability of a loss of decay heat removal by implementing procedure and hardware modifications to allow manual alignment of the fire protection system or by installing a component cooling water cross-tie.	$250K (procedure enhancement and minor mod) >$2M for new pumps	Assume $200K for minor modification and $50K for procedure change (both per site). Could also include installing additional SW pump(s) per Phase 1 SAMA #73	Screened. Procedure already in place to X-tie to opposite unit HPSW pumps; this is included in the model, but not credited. Small effect on CDF. A X-tie to FPS would not provide required flow. Cost for new hardware addition is >$2 million.	*A cost-benefit analysis using the 95th percentile CDF was performed. In this case, all HPSW component failure rates were set to zero. This left only human error terms as system failure contributions. Net value of –$44K (assuming only a $250K implementation cost) indicates that the SAMA is still not beneficial.*
3	Install an independent method of suppression pool cooling.	SAMA would decrease the probability of loss of containment heat removal.	>$2M	[>$1M/Unit x 2] NUREG-1437 cost for independent Containment Spray System is >$1M.	Screened ($)	*Maximum benefit would be to eliminate all loss of containment heat removal sequences (i.e., about 20% of CDF). This would roughly correspond to a $2.7M averted cost risk using the 95th percentile. This benefit is less than a realistic implementation cost (>>$2M) for a completely independent suppression pool cooling system for both units, and therefore is still not beneficial.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
4	Install a filtered containment vent to remove decay heat.	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products being scrubbed. Option 1: Gravel Bed Filter Option 2: Multiple Venturi Scrubber	>$2M	[$3M/Unit X 2] - Ref. 17, Section A.5.5.1	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
5	Install a containment vent large enough to remove ATWS decay heat.	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	>$2M	[$300K/Unit x 2] - Ref. 17, Section A.5.11.1, but installation of hard pipe vent at PB cost >$2 million (Ref. 18)	Screened ($)	*Maximum benefit would be to eliminate all ATWS scenarios (i.e., about 10% of CDF). This would roughly correspond to a $1.35M averted cost risk using the 95th percentile base cost risk of $13.5M. This is less than the estimated implementation cost (>$2M per unit), and therefore is still not beneficial. Also note that the current hard-piped vent is 16" in diameter and is capable of removing decay heat in a ll but the most severe full ATWS cases.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
6	Use the fire protection system as a backup source for the containment spray system.	SAMA would provide redundant containment spray function without the cost of installing a new system.	$50K	[$25K/Unit x 2] - Hatch Submittal, Section 5.1 Also consider as a fire protection as a means for low pressure injection per Phase 1 SAMA #154	Screened. Hardware failure of containment spray is not a factor in the system evaluation. The drywell spray initiation limit defined by the EOPs prevents its use in the cases where it would potentially provide benefit (flooding the drywell floor prior to vessel failure). Introducing an additional source of water to the CS system will not affect the model's quantification. No detailed analysis required.	*N/A – also see response to RAI #6.*
7	Install a passive containment spray system.	SAMA would provide redundant containment spray method without high cost.	>$2M	Assumed to be similar in cost to passive HP system (SAMA 149)	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
8	Construct a building to be connected to primary/secondary containment that is maintained at a vacuum.	SAMA would provide a method to depressurize containment and reduce fission product release.	>$2M	$'s per engineering judgment	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
9	Proceduralize alignment of spare diesel to shutdown board after loss of offsite power and failure of the diesel normally supplying it.	SAMA would reduce the SBO frequency.	See SAMA 56	Need to install spare D/G to benefit from this SAMA. Spare DG is screened based on cost (See SAMA 56)	Screened ($)	*See disposition for Phase II SAMA 10 below.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
10	Provide an additional diesel generator.	SAMA would increase the reliability and availability of onsite emergency AC power sources.	>$2M	$'s per engineering judgment. Ref 17 lists cost at approximately $1.2M. However, this is significantly less than cost of installing new DGs after plant is built (Calvert Cliffs >$100M for 2 new DGs).	Screened ($)	*Maximum benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. Therefore, it can be approximated that the maximum benefit using the 95th Percentile CDF value would be about 6.75 times that value, or $1.9M. Based on estimated implementation costs of >>$2M, the SAMA is still not beneficial.*
11	Provide additional DC battery capacity.	SAMA would ensure longer battery capability during an SBO, reducing the frequency of long-term SBO sequences.	$1.6M	Assume $200K/battery x 8 batteries (includes analysis, equipment, and modification implementation)	Detailed cost-benefit analysis performed. Net value of -$1,334,903 indicates that this modification is not beneficial. Refer to section G.5.2.	*Revised cost-benefit analysis using the 95th percentile CDF was performed. Net value of +$148K indicates that the SAMA may be beneficial. (See discussion below.)*
12	Use fuel cells instead of lead-acid batteries.	SAMA would extend DC power availability in an SBO.	>$2M	[$6M] - Ref. 17, Section A.5.10.1	Screened ($)	*Maximum benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. Therefore, it can be approximated that the maximum benefit using the 95th Percentile CDF value would be about 6.75 times that value, or $1.9M. Based on estimated implementation costs of >>$2M, the SAMA is still not beneficial.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	**SAMA title**	**Result of potential enhancement**	**Estimated Cost**	**Comment**	**Original Phase II Disposition**	***Revised Phase II Disposition***
13	Develop procedures to repair or replace failed 4-kV breakers.	SAMA would offer a recovery path from a failure of the breakers that perform transfer of 4.16-kV nonemergency busses from unit station service transformers, leading to loss of emergency AC power.	$50K	Assume $50K for site procedure change	Detailed cost-benefit analysis performed. Net value of -$49,612 indicates that the SAMA is not beneficial. Refer to section G.5.3.	*Revised cost-benefit analysis using the 95th percentile CDF was performed. Net value of -$48.0K indicates that the SAMA is still not beneficial.*
14	Install gas turbine generator.	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	>$2M	$'s per engineering judgment	Screened ($)	*Maximum benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. Therefore, it can be approximated that the maximum benefit using the 95th Percentile CDF value would be about 6.75 times that value, or $1.9M. Based on estimated implementation costs of >>$2M, the SAMA is still not beneficial.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	**SAMA title**	**Result of potential enhancement**	**Estimated Cost**	**Comment**	**Original Phase II Disposition**	***Revised Phase II Disposition***
15	Proceduralize intermittent operation of HPCI.	SAMA would allow for extended duration of HPCI availability.	\$50K	Hatch estimate is \$22,200/unit (Section 5.2). Assume \$50K for site procedure change at PBAPS.	Screened. Intermittent operation of HPIC for SBO cases is detrimental to battery life and is judged not to be desirable. For LOOP cases, room cooling was determined not to be required (ECR 96-00367) for operation of HPCI; however, procedures already exist to align alternate room cooling for extended operation should the need arise and are considered more appropriate than multiple turbine restarts. It should also be noted that RCIC is preferred if both systems are available during LOOP and HPCI would potentially be terminated by 10 minutes after trip (per SE-11 bases, section B-6).	*Use of the 95th Percentile CDF does not change the disposition.*
16	Install motor-driven feedwater pump.	SAMA would increase the availability of injection subsequent to MSIV closure.	>\$2M	\$'s per engineering judgment	Screened (\$)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
17	Enhance procedure to instruct operators to trip unneeded RHR/CS pumps on loss of room ventilation.	SAMA increases availability of required RHR/CS pumps. Reduction in room heat load allows continued operation of required RHR/CS pumps, when room cooling is lost.	$50K	Assume $50K for site procedure change	Screened. The largest Risk Reduction Worth associated with CS, LPCI, and NSW, including common cause failures is 1.003. This indicates that no significant change to the PSA will occur if the room cooling dependency is improved or removed from the model; thus, a positive net value is not achievable. No detailed analysis is required.	*Use of the 95th Percentile CDF does not change the disposition.*
18	Increase the safety relief valve (SRV) reseat reliability.	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after standby liquid control (SLC) injection.	$2M	Assume $200K/SRV x 10 ADS SRVs (5 per site) plus additional 12 non-ADS SRVs. This includes analysis, equipment (assumes replacing SRVs with new models) and modification implementation.	Detailed cost-benefit analysis performed. Net values of -$1,906,215 (Case A) and -$1,825,762 (Case B) indicate that the SAMA is not beneficial. Refer to section G.5.4.	*Revised cost-benefit analysis using the 95th percentile CDF was performed. Net value of -$824K indicates that the SAMA is still not beneficial.*
19	Modify Reactor Water Cleanup (RWCU) for use as a decay heat removal system and proceduralize use.	SAMA would provide an additional source of decay heat removal.	>$2 million for hardware upgrade	Proceduralizing the use of RWCU as a decay heat removal system could be cost-effective. However, RWCU heat removal capacity may be low.	Screened. The PBAPS RWCU system is incapable of serving as the sole DHR system until many days after shutdown and therefore is virtually ineffective for accidents at full power. No detailed analysis required.	*Use of the 95th Percentile CDF does not change the disposition.*

TABLE 3-1 Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value						
Phase II SAMA ID number	**SAMA title**	**Result of potential enhancement**	**Estimated Cost**	**Comment**	**Original Phase II Disposition**	***Revised Phase II Disposition***
20	2.a. Passive High Pressure System	SAMA will improve prevention of core melt sequences by providing additional high presssure capability to remove decay heat through an isolation condenser type system	>$2M	[$1.7M x 2] - Ref. 17, Section A.5.2.1	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
21	2.c. Suppression Pool Jockey Pump	SAMA will improve prevention of core melt sequences by providing a small makeup pump to provide low pressure decay heat removal from the RPV using the suppression pool as a source of water.	$480K	Ref. 17, Section A.5.2.3 lists cost as $120K (per unit). However, since this is for a plant not yet built, estimate a factor of 2 more cost for PBAPS. Therefore, cost is $120K/unit x 2 Units x 2 = $480K	Detailed cost-benefit analysis performed. Net value of -$129,044 indicates that the SAMA is not beneficial. Refer to section G.5.5.	*Revised cost-benefit analysis using the 95th percentile CDF was performed. A net value of +$1.85M is obtained using the same optimistic PSA assumptions that were used in the original analysis. This indicates that the SAMA may be beneficial. (See discussion below, however, that indicates that a more realistic PSA analysis would indicate that this SAMA is still not beneficial.)*
22	2.e. Additional Active High Pressure System	SAMA will improve reliability of high pressure decay heat removal by adding an additional system.	>$2M	Assumed to be similar in cost to passive HP system (SAMA 149)	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
23	2.h. Safety Related Condensate Storage Tank	SAMA will improve availability of CST following a Seismic event	>$2M	[>$1M x 2] - Ref. 17, Section A.5.2.4	Screened ($)	*It is judged that a minimal amount of benefit would be obtained from installing a seismically qualified CST. Therefore, the costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
24	3.c. Improved Vacuum Breakers (redundant valves in each line)	SAMA reduces the probability of a stuck open vacuum breaker.	>$2M	$'s per engineering judgment. ABWR, Section 5.3.3 lists cost as >$100K (per unit). However, this is for a plant not yet built. This is an extensive modification, so cost is estimated at >$1M/unit.	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
25	8.e. Improved MSIV Design		>$2M	Assume $200K/MSIV x 16 MSIVs (8 per unit)	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
26	9.a. Steam Driven Turbine Generator	This SAMA would provide a steam driven turbine generator which uses reactor steam and exhausts to the suppression pool. If large enough, it could provide power to additional equipment.	>$2M	[$6M x 2} - Ref. 17, Section A.5.9.1	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if using the 95th percentile CDF.*
27	9.f. Improved Uninterruptable Power Supplies	SAMA would provide increased reliability of power supplies supporting front-line equipment, thus reducing core damage and release frequencies.			Screened. The UPSs are not included in the PBAPA PSA and are not considered to be risk significant; thus, it is not possible to obtain a positive net value with this SAMA. No detailed analysis required.	*Use of the 95th Percentile CDF does not change the disposition.*
28	9.i. Dedicated RHR (bunkered) Power Supply		>$2M	[$1.2M x 2] - Ref. 17, Section A.5.9.2	Screened ($)	*Maximum benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. Therefore, it can be approximated that the maximum benefit using the 95th Percentile CDF value would be about 6.75 times that value, or $1.9M. Based on estimated implementation costs of >$2M, the SAMA is still not beneficial.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
29	10.a. Dedicated DC Power Supply	This SAMA addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components (e.g., RCIC).	>$2M	[$3M x 2] - Ref. 17, Section A.5.10.1	Screened ($)	*<u>Maximum</u> benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. Therefore, it can be approximated that the maximum benefit using the 95th Percentile CDF value would be about 6.75 times that value, or $1.9M. Based on estimated implementation costs of >>$2M, the SAMA is still not beneficial.*

TABLE 3-1
Sensitivity of Phase II Dispositions Using 95th Percentile CDF Value

Phase II SAMA ID number	SAMA title	Result of potential enhancement	Estimated Cost	Comment	Original Phase II Disposition	*Revised Phase II Disposition*
30	10.d. DC Cross-ties	This SAMA would improve DC power reliability.	$250K	Assume $200K for minor modification, plus $50K for procedure change. Only partially addressed by SAMA 61	Screened. The PBAPS SE-11 procedure has been developed to optimize cross-tie capabilities of the 4 kV buses and various power supplies afforded by the emergency diesel generators and the dedicated offsite power source from Conowingo Dam. One of the main tenets of this procedure is to ensure that 4 kV power is available to all necessary DC bus chargers. It is judged that adding DC cross-tie capabilities would not be cost effective since the optimum benefit is already obtained from the SE-11 procedure. The DC buses and batteries are very reliable, and providing 4 kV power to the battery chargers is the most beneficial way of ensuring that DC power remains available.	*Use of the 95th Percentile CDF does not change the disposition.*

1

Summary Response to RAI 3a

In summary, it was found that the Phase II disposition of two of the SAMAs would potentially be different if the screening were based on the upper bound of the CDF distribution rather than the mean value. These two situations are each described in more detail below.

Phase II SAMA ID Number 11 (Provide additional DC battery capacity)

The updated cost-benefit analysis using the 95th percentile CDF with the same PSA model assumptions that were used in the original analysis (i.e., that extended battery life assumptions from 2 to 4 hours) indicates that the total averted cost risk would be approximately $1.75M. When compared to an estimated implementation cost of $1.6M, this results in a potential cost-benefit of about $150K. However, given that the more realistic cost estimate for replacing eight battery divisions between the two units is most likely to be much greater than $2M, and since the $150K of potential averted cost is only obtained when the 95th percentile CDF is used as the cost basis, then it is judged that this SAMA is still not cost-beneficial.

Phase II SAMA ID Number 21 (Add suppression pool jockey pump)

The updated cost-benefit analysis using the 95th percentile CDF with the same PSA model assumptions used in the original analysis (i.e., that a totally independent system could be used to provide long-term injection to the RPV with an optimistic reliability value of 0.01) indicates that the total averted cost risk would be approximately $2.33M. When compared to an estimated implementation cost of $480K, this results in a potential cost-benefit of about $1.85M. This benefit, however, is highly skewed by the optimistic PSA model assumptions used in the original analysis, and by the underestimated costs that would be associated with a completely independent system.

A more realistic PSA calculation was therefore performed which assumes that the jockey pump is supplied by the E2 480V bus (i.e., the bus with the lowest current risk achievement worth in the model), and consists of a total system reliability of 0.05 (e.g., including human error contribution) instead of an optimistic value of 0.01. In this case, the total averted cost is approximately $540K. When compared to an estimated implementation cost of $480K, this results in a potential cost-benefit of about $60K. However, since this cost is judged to be underestimated (even for a system that is not totally independent from other systems at the site), and since the $60K of potential averted cost is only obtained when the 95th percentile CDF is used as the cost basis, then it is judged that this SAMA is still not beneficial.

Response to RAI 3b

It is acknowledged that the methods used for the Peach Bottom IPEEE do not provide numerical estimates of the CDF contributions from seismic and fire initiators. However, it is believed that the current risk associated with external events is much lower than that which existed at the time of publication of NUREG/CR-4551 because many plant improvements have been made since that time, mostly as a result of the insights obtained from the IPEEE for Peach Bottom. These improvements include:

- Increased Fire brigade awareness of important fire areas.
- Incorporation of automatic sprinklers in 4 kV switchgear areas.
- Incorporation of sprinklers in the 13 kV area and the addition of the sprinkler heads on the 116' elevation leaving the 13 kV area into the remainder of the turbine building, (i.e., creating a water curtain at the openings.)
- Thermo-lag replacement and upgrade in several fire areas.
- Miscellaneous equipment replacement and/or upgrade for resolution of Generic Safety Issue A-46,"Seismic Qualification of Equipment in Operating Plants".

Additionally, if the estimate of the contribution from external events from NUREG/CR-4551 had been incorporated into the evaluation, the increase would have been less than that provided by the 95th percentile CDF estimate from internal events. Consequently, the sensitivity of the results and screening process if using the 95th percentile CDF value as described in the response to 3a above provides a bounding assessment of the potential impact from including the risk from external events.

Response to RAI 3c

As described in the response to 3a above, SAMAs 11 and 21 would potentially be beneficial if the screening were conservatively based on the upper bound of the CDF rather than on the mean value. However, in both cases, a more realistic cost estimate or more realistic credit in the PSA analysis indicates that the SAMAs are not beneficial.

Response to RAI 3d

The implementation costs and net values reported in Section 4.20.6, Table G.4-2 (and Table 3-1 above) are based on the combined value for both units, and as such are consistent with the baseline costs of a severe accident as used in the screening process.

4. ***On page E.4-41, the submittal states that the Peach Bottom generating capacity has increased from 3293 to 3458 MW(t), i.e., by 5 percent. It is recognized that the effect of the power uprate on the Level 1 PSA success criteria has been considered, and that the impact on the initial core radiological inventory used in the SAMA analysis has not been taken into consideration. However, the submittal does not provide any information on the effect of power uprate on accident progression as modeled in the Level 2 PSA, e.g., the impact on the timing and depressurization capability of containment venting. Please indicate if the impact of the power uprate on containment failure/release times, and the magnitude of radiological releases to the environment has been considered. If this is not the case, please provide justification for neglecting this impact.***

Response to RAI 4

The Peach Bottom PSA has a detailed Level 2 PSA model that allows the calculation of radionuclide release frequencies distributed in 16 end states. The detailed Level 2 PSA model and its documentation were reviewed to identify those areas of the model that could be influenced by the 5% power uprate and may therefore influence the SAMA evaluation.

As noted, the total radionuclide inventory can be considered to increase due to the 5% power uprate. As a first approximation, this translates into a 5% increase in the radionuclide inventory. (Certain radioisotopes will change by more or less than this depending on burnup.)

For the remainder of this discussion, the focus is on the power uprate impact on changing the percent radionuclide release or its frequency.

PSA uncertainties between the median and the 5% or 95% bounds are generally estimated at 500% or larger. Therefore, relatively small perturbations of 5% should be negligible in assessing the propriety of plant modifications unless step changes in success criteria, timing, or releases occur as a result of a perturbation.

The binning of accident sequences into discrete release categories represents a method that has been used in virtually all PSAs, starting with WASH-1400. This technique characterizes the spectrum of releases into discrete bins. For the purposes of SAMA and this RAI, it is judged that perturbations of the PSA that do not move radionuclide releases from one bin to another and do not significantly change the release frequency (i.e., less than 5%) can be considered negligible effects on the SAMA conclusions. The conclusions from the Level 2 review and evaluation can be summarized in terms of major topic areas related to functional response as usually modeled in a Containment Event Tree. Table 4-1 summarizes this discussion.

TABLE 4-1
Level 2 Functional Response Effect due to 5% Power Uprate

Functional Response	5% Power Uprate Influence	Effect on SAMA
Containment Isolation	The containment isolation failure probability is determined to be dominated by pre-existing incipient failure modes. No correlation between these failure modes and the 5% power uprate were identified.	None
RPV Depressurization	The ability to depressurize the RPV in postulated accident sequences to allow in-vessel recovery of a damaged core or to prevent RPV breach at elevated pressure was explicitly examined because of their importance in determining radionuclide releases and frequencies. The success criteria for depressurization is 2 SRVs out of the full complement of SRVs. This success criteria was determined to not be affected by the 5% power uprate.	None
	The timing associated with the depressurization decision is impacted. The timing estimates use the MAAP 3.0B RPV breach failure mode. Credit for in-vessel recovery is hindered by weaknesses in available core melt progression tools and has led to conservatively identifying the time to RPV breach. A change in time available for effective depressurization has been chosen to be from 40 minutes for the plant configuration before 5% power uprate to 38 minutes after based on MAAP 3.0B. Other core melt progression tools (e.g., MELCOR, MAAP4) indicate substantially longer times are available for in-vessel recovery.	Negligible
	This could translate into small increases (5E-06 to 5E-05) in the Human Error Probability (HEP) for this action; however, the conditional probability of the HEP is already 0.5 in the Level 2 model.	Negligible
	Therefore, there is no measurable impact of the 5% power uprate on the functional event of RPV depressurization.	Negligible

TABLE 4-1
Level 2 Functional Response Effect due to 5% Power Uprate

Functional Response	5% Power Uprate Influence	Effect on SAMA
In-Vessel Recovery	This function is similar to the assessment of RPV depressurization in that there are two aspects: • the success criteria • the timing of actions The success criteria used in the PSA considers entire large volume subsystems or trains required for successful in-vessel recovery. Those include: CS, LPCI, FPS, HPSW cross-ties. These systems can provide significantly more water than required to cool the core and restore RPV water level above TAF for all sequences including core damage events (except excessive LOCA or large DBA LOCA). Therefore, no change in success criteria is found. CRD and SLC are not credited with the capability to be adequate for in-vessel recovery.	None
	The timing change is found to be similar to RPV depressurization and the numerical differences to not be measurable in the calculation of the HEP for these small changes in time available for action.	Negligible
Energetic Phenomena	One of the critical aspects of the Level 2 analysis is the assessment of phenomenological effects that can cause RPV and/or containment failure in an energetic manner. These phenomena include: • In-vessel steam explosion • Ex-vessel steam explosion • Direct Containment Heating (DCH) • Missiles • Cryogenic induced pre-existing failures • Pedestal failure • Recriticality • Vapor suppression failure • Hydrogen deflagration (during deinerted operation) Each of these effects is quantified in the Peach Bottom Level 2 evaluation. It is found that these energetic phenomena are governed by core melt progression characteristics that are considered to be of low probability. No direct correlation could be identified between the 5% power uprate and the change in likelihood of these phenomena.	None

Ex-Vessel Debris Cooling	The assessment of ex-vessel debris cooling is a critical aspect of the Level 2 assessment for Mark I containments. This assessment includes: • The availability of water • The procedural directions and capability to achieve the water injection • The phenomena of drywell shell melt-through that can cause the containment to fail and result in substantial and "early" radionuclide releases The availability of water is assessed given the pre-conditions that exist that have led to core damage and RPV breach, i.e., failure to deliver water sources to the core. This means very few options are available at this time. One of the remaining options considered is the use of the drywell spray path to provide water on the drywell floor and subsequent debris cooling. The procedural direction for the water injection to the containment via drywell sprays has recently been substantially improved using the latest EP/SAG revisions to virtually guarantee DW sprays to cool debris, if the system is available. Again, the HEP change associated with a 2 minute time change over a 40-minute assessment period[1] is considered negligible. Therefore, a negligible impact on the 5% power uprate results. Finally, the DW shell is always assumed to fail if water cannot be delivered to the debris. The 5% power uprate does not have any additional adverse impact on this calculation of DW shell failure probability.	None Negligible None
Containment Flooding	Another response to core melt progression that occurs is the entry of the operating crew into the SAMGs and the direction to flood containment. The current EP/SAGs and Peach Bottom Transient Response Implementation Procedures (TRIPs) are significantly improved with respect to containment flooding from the older Rev 4-based TRIPs. The containment flooding evolution is severely restricted to minimize adverse impacts on the public. The 5% power uprate is not considered to measurably influence the procedural direction, timing, fraction of release, or perception of the crew.	None

[1] The 40-minute assessment period is the time from the release of radionuclides to the containment, i.e., core damage, to the time that RPV breach might occur. The 40 minutes is judged to be conservative. However, if the time available for DW spray initiation is extended, the net result is to further reduce the change in HEP associated with this action.

Containment Venting	Of the multiple core melt progression scenarios, one of these sets relates to successful debris control either in-vessel or ex-vessel with the containment initially intact. A remaining critical safety function is the pressure control for containment. Containment venting is one method to satisfy the pressure control function. The time at which containment venting is needed can range from approximately 20 hours for non-ATWS[(1)] in-vessel recovery scenarios to 8-10 hours for ex-vessel recovery scenarios. The 5% power uprate can alter these times by 1 hour to 24 minutes, respectively. For actions with this amount of time available, the failure probability and the HEP are dominated by other considerations.	None
Containment Failure Mode	The containment failure mode is a critical parameter in the assessment of the radionuclide release fraction and, therefore, the frequency of the associated bins. However, no direct correlation has been identified with the 5% power uprate. The containment failure modes include both location (drywell, wetwell air space, wetwell water space) and size. These failure modes are determined based on the containment conditions (pressures, temperatures, and dynamic loads). These conditions can be influenced by the initial power level; however, the calculated changes in these parameters is relatively small compared with the core melt progression effects. While the time to containment failure can be influenced by the change in power level, the probability of a failure location and size is much less sensitive and for all intents and purposes is unaffected by the small 5% power uprate. Therefore, no measurable impact is found on the Level 2 results.	None
Reactor Building Effectiveness	A passive measure that can contribute to a reduction in the radionuclide release is the ability of the Reactor Building to act as a radionuclide retention location. The 5% power uprate may result in increased hydrogen generation which has adverse impacts on the Reactor Building effectiveness as a radionuclide mitigator due to burning in the secondary containment. Because the Reactor Building has only a small potential benefit (e.g., generally a ~5% probability of effectiveness), the 5% power uprate is not considered to significantly alter this small probability of success, i.e., it cannot decrease below 0% effectiveness.	Yes, but not significant

(1) ATWS scenarios that result in containment failure and core damage do not require containment venting.

Summary Response to RAI 4

In summary, the detailed Level 2 for Peach Bottom was examined for areas where the 5% power uprate could result in increases in radionuclide release magnitudes or frequencies. No step change in magnitude, frequency, or timing of radionuclide release was identified. Small potential variations in operator action times, accident timing, or in Reactor Building effectiveness were identified. These small changes are not considered distinguishable within the uncertainties present in the Level 2 PSA. For example, none of the changes are judged to result in moving sequences from one radionuclide release bin to another.

5. ***Based on information provided in Section 4.20.5, the SAMA candidates for Peach Bottom were developed from lists of SAMAs considered at other plants, NRC documents, and documents related to the advanced boiling water reactor. No mention is made of whether/how the plant-specific risk study was used to identify candidate SAMAs, such as performing a systematic examination of the top cutsets and leading contributors to large release, or conducting basic event importance analyses/rankings to identify candidate SAMAs. In this regard:***

   a. ***Please provide a description of how the plant-specific risk study was used to identify candidate SAMAs,***

   b. ***If a systematic examination or importance analysis based on the plant-specific PSA was not performed, please justify why the approach utilized in the submittal is sufficient to identify all potentially cost-beneficial SAMAs aimed at reducing CDF and risk dominant releases, and***

   c. ***Provide a copy of the "PBAPS Report on Accident Management Insights" referred to in Note 14 to Table G.4-1.***

Response to RAI 5a

Severe accident mitigation alternatives have been investigated by a large number of utilities and NRC contractors. The generic SAMA list compiled by Exelon includes items identified as potential risk-reducing techniques based on industry experience and insights. The compiled list of SAMAs to examine for Peach Bottom made use of this extensive effort by including the results of this effort in the SAMA review. However, it was also recognized that this was not sufficient by itself and that plant specific analyses should be used to supplement the generic information to establish whether there were additional plant or procedural modifications that should be examined. The following sources were included in the determination of the complete SAMA list for Peach Bottom:

- The Peach Bottom IPE implemented a systematic process for reviewing important plant systems, structures, and components (SSCs) to identify potential Accident Mitigation Strategies (AMSs). This was performed in addition to addressing a list of generic AMSs from supplement 2 of generic letter 88-20. As the list of important SSCs is essentially the same for the current PSA model as it was for the IPE, the plant specific SAMA review is judged to be addressed by the IPE. Those AMSs that were identified in the IPE as beneficial in reducing risk in a measurable manner and applicable to the Peach Bottom site have already been implemented. These include an enhanced version of procedure SE-11, "Loss of Offsite Power," and the Torus Hard Piped Vent.

- Upon completion of the periodic updates of the PSA model (e.g., in 1997 and 1999), the results were carefully examined to determine how the risk profile and list of important systems has changed from the previous results. A significant change in the results has not occurred as can be noted in the response to RAI 1 above.

- The Peach Bottom PSA was reviewed to identify those potential contributors that made a significant contribution to core damage frequency or risk. These contributors were seen as candidates for further investigation regarding cost beneficial modifications to reduce risk. It was confirmed that the SAMA list used in the original submittal included possible alternatives to the most important systems at Peach Bottom (i.e., HPCI, RCIC, RHR, the Emergency Diesel Generators, AC Power, DC Power, and miscellaneous service water systems).

- In addition, a plant-specific risk management insights document was developed as part of accident management implementation that provided plant specific insights that could enhance safety. These insights were also made part of the SAMA list.

Therefore, it is judged that a systematic and thorough process using both industry and plant specific inputs were used in compiling the list of SAMAs to investigate.

Response to RAI 5b

See response to RAI 5a.

Response to RAI 5c

A copy of the report is attached.

6. ***Table G.4-1 states that there are procedures in place at Peach Bottom which allow containment flooding (see Phase I SAMA numbers 41 and 51). Please provide the following information:***

   a. ***A description of the version of the BWROG Emergency Procedure and Severe Accident Guidelines (EP/SAG) that are currently in place at Peach Bottom, and that are credited in the PSA,***

   b. ***A discussion of how containment flooding would be accomplished at Peach Bottom during SBO events,***

   c. ***A description of how the impact of containment flooding has been considered in the Level 2 PSA (e.g., its impact on liner melt-through, wetwell vent submergence, and containment venting). If flooding is accounted for in the PSA, please explain why collapsed bins #3 and #4 that are primarily due to "drywell shell melt-through" account for a major portion of the population risk in Table G.2-3, and***

   d. ***Additional justification for dispositioning Phase II SAMA number 6 in Table G.4-2 (which states that "the drywell spray initiation limit defined by the EOPs prevents its use in the cases where it would potentially provide benefit"), given that Revision 2 of the EP/SAG has relaxed the drywell spray initiation limit to permit the use of drywell sprays under a broader set of conditions.***

Response to RAI 6a

The BWROG EP/SAG Revision 1 procedures are implemented at Peach Bottom. The human actions associated with the PSA model used to evaluate SAMA are based on both EP/SAG Revision 1 and EPG Revision 4 guidance. The current Level 1 PSA model incorporated an update of key human actions using guidance and procedures derived from EP/SAG Revision 1. Priority was placed on those human actions that were identified as important in the BWROG peer certification comments. The impact on the Level 2 analysis is described in the response to 6c and 6d below.

Response to RAI 6b

Containment flooding is directed using the Severe Accident Management Procedure (SAMP-1) and is addressed somewhat differently given specific RPV and containment conditions.

A Station Blackout (SBO) represents a small contributor to the CDF compared to all loss of offsite power initiated events. The incorporation of the Conowingo tie-line and the associated cross-tie procedures have reduced the frequency associated with SBO at Peach Bottom. The dominant contributor to CDF at PBAPS is a LOOP where one or more diesels are available but not cross-tied to supply sufficient functions needed to prevent core damage. These LOOP situations, should the containment and RPV conditions warrant, would utilize the emergency AC-backed systems of RHR and High Pressure Service Water (HPSW) for drywell spray and containment flooding. An SBO presents a special case in that only DC-backed systems and the diesel fire pump would be available for RPV or containment injection. Although procedures exist to align the diesel fire pump through the RHR system for RPV injection, little credit is given for its use. No specific procedure exists to spray or flood the containment using the diesel fire pump during an SBO. (A more detailed explanation is provided in the responses to 6c and 6d below.) In any event, the discharge pressure, flow restrictions, RPV and containment backpressure, and elevation differences all contribute to limit the benefit of the diesel fire pump in any containment flooding scenarios.

Response to RAI 6c

The Level 2 PSA explicitly considers the impact of containment flooding.

The following discussion provides the impact of containment flooding for two situations:

- Case A – Use of EPG, Rev. 4 as currently modeled in the Level 2 PSA
- Case B – Use of EP/SAG, Rev. 1

Before proceeding, however, it should be noted that in either case, containment flooding does not prevent drywell shell melt-through. This is because the timing associated with the core melt progression and debris attack of the drywell shell is shorter than the time required to flood the containment by the filling the torus to above the drywell floor.

CASE A

The consideration of containment flooding for the Peach Bottom Level 2 PSA is addressed for the following items for the TRIPs based on EPG Rev 4:

Drywell Shell Melt-Through: Given a core melt progression that prevents restoration of cooling above top of active fuel (TAF), the EPGs prescribe containment flooding to cover the debris. However, containment flooding does not prevent drywell shell melt-through, because the timing associated with the core melt progression and debris attack of the drywell shell is shorter than the time required to flood the containment above the drywell floor. Therefore, shell melt-through is not mitigated by "containment flooding."

It is noted that aspects of the containment flood strategy have side benefits of debris coolability under certain conditions such as post-RPV breach. It is also noted that despite procedural direction to implement "containment flooding," if the systems were unavailable to prevent core damage, there is a high conditional probability that they are unavailable to prevent drywell shell failure. Hence, while drywell shell failure can be reduced in probability by having a procedure, it likely cannot be eliminated as a failure mode.

Drywell Sprays: The use of drywell sprays in conjunction with containment flooding during a core melt progression event is severely limited by the Drywell Spray Initiation Limit (DWSIL) curve and the direction regarding initiation. While the use of drywell sprays could prevent shell failure, the restrictions imposed by the above two were such that the EPG, Rev. 4 based TRIPs all but preclude the use of the drywell sprays.

Containment Venting: The wetwell containment vent is from the torus airspace. If and when containment flooding occurs, the wetwell vent will be precluded from operation. This results in the need for use of the drywell vent if containment flooding is being used. This is explicitly accounted for in the Level 2. High early releases are assigned to the containment flood process due to the operation of the drywell vent.

Downcomer Submergence: The downcomers and vents from the drywell to the suppression pool provide the pathways for steam and fission products in the drywell to reach the suppression pool. Containment flooding tends to increase the downcomer pipe submergence. The EPG Rev 4 based TRIPS do not limit the degree of submergence prior to RPV breach. Vapor suppression is likely compromised prior to RPV breach. This adverse impact results in immediate and energetic containment failure at the time of RPV breach (either at high or low pressure).

This is modeled in the Level 2 PSA.

RPV Venting: The containment flood direction from EPG Rev. 4 guaranteed RPV venting would be implemented if containment flooding were successful. RPV venting was found to have severely adverse impacts on radionuclide release, i.e., a LERF contributor. This Peach Bottom PSA insight was one of the inputs to the EP/SAG revision which then substantially restricted when RPV venting would be allowed.

Next, consider the treatment in Level 2 when the EP/SAG generic procedures are implemented.

Case B

Containment flooding using the EP/SAG directions has significantly altered the directions to the operating crew. Containment flooding is allowed in only a narrow window of symptoms. In lieu of the "old" containment flooding procedure (i.e., based on EPG, Rev. 4), the "new" procedure (i.e., based on EP/SAG, Rev. 1) is much more selective in its implementation of filling the containment with water. However, the "new" guidance is also much more proficient at identifying the ability to cool debris using the drywell sprays.

The consideration of containment flooding is addressed for the following items for the updated TRIPS based on EP/SAG guidance:

Drywell Shell Melt-Through: Given a core melt progression that is projected to breach the RPV, and prior to RPV breach, the EP/SAGs do not prescribe containment flooding to cover the debris. Therefore, shell melt-through is not mitigated by "containment flooding." However, the EP/SAGs have supplemented the prohibition against containment flooding with a substantial improvement in the direction to spray the drywell and get water on the drywell floor.

The net effect is to significantly delay or preclude altogether drywell shell failure due to debris attack of the shell when systems are available to support effective drywell spraying.

Drywell Sprays: The strategy for SAMGs makes improved use of drywell sprays to further attempt to provide debris cooling, Radionuclide scrubbing, and increased pressure suppression capability. This strategy is manifested in: (1) improved directions to initiate drywell sprays; and, (2) relaxation in the restrictions on the Drywell Spray Initiation Limit (DWSIL) curve. These changes from EPG, Rev. 4 enhance the potential for debris cooling and prevention of shell melt-through.

Containment Venting: The wetwell containment vent is from the torus airspace. If and when containment flooding occurs, the wetwell vent will eventually be precluded from operation. This results in the need for use of the drywell vent if containment flooding is being used. This is explicitly accounted for in the Level 2 PSA.

Downcomer Submergence: The downcomers and vents from the drywell provide the pathways for steam and fission products in the drywell to reach the suppression pool. Containment flooding tends to increase the downcomer pipe submergence. The EP/SAGs limit the degree of submergence prior to RPV breach to ensure that vapor suppression is not compromised. This decision is made in the EP/SAGs to preserve the containment integrity as long as possible by precluding an energetic containment failure mode at RPV breach.

RPV Vent: Severe restrictions are in place to prevent the use of RPV venting in cases where its potential benefit is negligible. An example is for the case with the RPV breached. This change in strategy results in a substantial reduction in release magnitude for certain severe accident scenarios with RPV breach and successful debris cooling.

Summary Response to RAI 6c

Containment flooding does not prevent drywell shell melt-through. The timing associated with the core melt progression and debris attack of the drywell shell is shorter than the time required to flood the containment above the drywell floor. As such, containment flooding is not designed in the current EP/SAGs to preclude drywell shell failure in the short term. This task has been relinquished to the use of drywell sprays for this purpose. In either case, the reality is that if a water source is available that can be used to flood containment or inject to the drywell sprays, it can, in all likelihood, get to the RPV to prevent core melt progression.

Response to RAI 6d

The SAMAs were dispositioned when the EPG Rev. 4 procedures were in place in the Peach Bottom Level 2 PSA. Therefore, the ability to use drywell sprays was severely restricted making a hardware modification ineffective. The technologies applied by the BWROG in the EP/SAGs and their upgrades have significantly relaxed the restrictions limiting the usefulness of drywell sprays, thereby making the use of drywell sprays viable for nearly all severe accidents for which the equipment can be made available.

The latest drywell spray initiation directions in the EP/SAGs provide improved guidance to support the use of drywell sprays for accident mitigation. These improved directions make the drywell spray usage a significant benefit for minimizing radionuclide releases. The primary benefit of DW sprays is associated with cases where an internal water source, i.e., RHR pumps, are available. A beneficial effect, albeit temporary, is also obtained from the use of external water sources to the drywell sprays.

The use of external water sources for DW sprays is allowed, at least for some time. The external water results in containment flooding as a by-product. As noted in Response to 6c, the latest EP/SAGs limit the water accumulation in containment if RPV breach has not been observed. As such, the external DW sprays will be terminated when torus level increases to near the ring header.

The EP/SAG application to the Peach Bottom TRIPS has resulted in using either RHR or HPSW for drywell spray. These two systems can be operated from the control room (if DC and emergency AC power are available) and supply more than sufficient water injection to make the DW sprays effective in preventing shell failure.

There are some accident sequences for which the AC power or DC power may not be available to operate RHR or HPSW. In such low frequency sequences, the use of a diesel fire pump could be considered as a water source to support drywell sprays. The Fussell-Vesely (F-V) importance for these sequences leading to core damage is approximately 0.1. This means that 10 % of CDF leading to possible radionuclide release could be influenced by the use of the fire protection system (FPS) for DW sprays.

The fire protection system can be procedurally aligned at Peach Bottom to the RHR system for RPV injection. There is not a procedure or guideline that directs this alignment to be used for the DW spray function of RHR. Based on the alignment procedure, the following is noted:

- The time to align could be quite long (e.g., hours)
- The flow rate to the RHR system is expected to be relatively low because of the following:
  - The diesel fire pump (DFP) head is 125 psig
  - There is a significant elevation differences between the pumps and the drywell spray discharge
  - There are significant head losses from the DFP to the discharge of the sprays
  - The containment back pressure under severe accident conditions which could be 10 - 100 psig depending on the accident scenario reduces the potential flow rate
- RHR and HPSW are the identified methods to use for DW sprays in the TRIPs/SAMGs
- The use of FPS would likely be relevant for cases where no power is available.

Summary Response to RAI 6d

In summary, FPS modifications are necessary to allow DW sprays to be beneficial as a DW spray source when other sources for DW sprays are expected to be unavailable. Either of two conditions would be needed:

Case A: Local operator action to manipulate the DW spray valves in the Reactor Building would be necessary because of the unavailability of AC power. This is judged not feasible because of the potential adverse radiation environment in the Reactor Building when the directive for DW sprays is given.

Case B: A modification to provide power that would be available in a site SBO to the DW spray valves. This latter modification is estimated at $0.5M[1][2]/plant or $1M/site.

[1] The basis for the plant modification is:

- Engineering for the power supply, cable, and interface with safety related equipment
- Procedure development

The key to understanding the additional impact of the FPS modification on the risk profile is the fact that if FPS is the only source to be available for DW sprays, i.e., for SBO sequences, then the ability to open the DW spray valves becomes the limiting aspect of the action.

The evaluation of the cost benefit associated with either Case A or Case B can be made as follows:

Case A: Modification to enhance the DFP flow rate into the RHR system ($50K/unit) results in no benefit because of the restriction on access to the Reactor Building for those accident sequences where its benefit could be seen, i.e., SBO conditions. Therefore, this is not cost beneficial.

Case B: Modification to enhance the flow rate and add supplemental power to the RHR injection valves is estimated at $0.5M/unit or a total of $1M/site. The benefit is calculated by assuming all SBO events can be successfully mitigated (conservative assumption). This leads to a maximum benefit of $284,000 using best estimate PSA calculations. This benefit is not sufficient to justify the projected costs on a best estimate basis.

Finally, Revision 2 of the BWROG EP/SAG, Appendix C: "Calculations", WS-3 "Drywell Spray Worksheet" has been revised making it less restrictive for high pressure and high temperature drywell conditions and to avoid possible restrictions on its use during situations where sprays are needed to submerge core debris. Revision 2 of the EP/SAG was issued by the BWROG in 2001 and has not yet been incorporated into PSA models.

- Hardware (portable AC generator)
- Spool piece from FPS to RHR system to achieve needed spray flow rate

(2) Estimated costs (in 1988 dollars) from NUREG/CR-5278 for this option are the following:

Mechanical:	Labor and Material	$466,800
	Engineering	71,300
Electrical:	Labor and Materials	66,400
	Engineering	9,100
		$613,600

We fully expect that Peach Bottom will adopt the revised calculation methodology and the resulting Drywell Spray Initiation Limit Curve because of the flexibility it will provide in initiating drywell sprays at elevated drywell temperatures and its use in SAMPs for core debris submergence. This will provide added benefit, but as described above, modifications to the existing system will not be cost beneficial.

7. ***In Table G.2-3, the population dose risk at 50 miles (last column) shows the total risk as 14.72 person-rem per reactor year. However, based on the values listed in the table, the total should be 2.10 person-rem per reactor year. It appears that the population dose for collapsed bin #4 is incorrectly shown as 1.28E-1 rather than 1.28E+1. Please confirm the correct population dose risk.***

Response to RAI 7

The correct population dose risk for collapsed bin #4 in Table G.2-3 is 1.28E+01. The calculation performed in support of the 1.28E+01 value was also checked to verify the correct bin #4 and total population dose risk. During reformatting of Table G.2-3 for the submittal document the exponent was inadvertently changed.

8. ***According to the Regulatory Analysis Guidelines (NUREG/BR-0058, Revision 2), sensitivity studies should be performed to assess the value of SAMAs over the license renewal period. The guidelines indicate that an alternative analysis using a 3 percent real discount rate should be prepared for sensitivity analysis purposes, and, as a general principle, additional sensitivity or uncertainty analyses, or both should be performed whenever the values of key attributes can vary widely. Such attributes could include core damage frequency, population and meteorology data, and evacuation assumptions. It is not apparent that such sensitivity studies were conducted. Please indicate if any sensitivity studies were conducted, and if so, what were the results of such studies.***

Response to RAI 8

A sensitivity study has been performed in order to identify how the conclusions of the SAMA analysis might change based on the value assigned to the real discount rate. The original real discount rate (RDR) of 7 percent has been changed to 3 percent and the maximum averted cost-risk was re-calculated. The Phase 2 coarse screening was re-examined using the revised maximum averted cost-risk to identify any SAMA candidates that could no longer be screened based on the premise that their costs of implementation exceeded all possible benefit. In addition, the most cost restrictive case identified in RAI 3 was examined to characterize the impact of the real discount rate on the detailed analyses that were performed.

The effect of implementing a 3 percent real discount rate on the maximum averted cost-risk compared with a 7 percent real discount rate is an increase of 33.5 percent. This results in a best estimate increase from $2.04M to $2.7M. Five SAMAs which were previously eliminated based on the coarse screening process were retained for further analysis as their costs of implementation were estimated to be less than the revised maximum averted cost-risk ($2.7 million). Table 8-1 summarizes the results of the detailed evaluations of Phase 2 SAMAs 3, 5, 23, 24, and 28. As the risk reduction associated with the implementation of these SAMAs is relatively small, application of the 3% real discount rate does not change the disposition of these SAMAs. The averted cost-risk for each of these SAMAs is well below the estimated cost of implementation and are, therefore, not cost beneficial.

TABLE 8-1
Sensitivity of Phase II Dispositions Using A 3% Real Discount Rate

Phase II SAMA ID number	**SAMA title**	**Result of potential enhancement**	**Estimated Cost**	**Comment**	**Original Phase II Disposition**	***Revised Phase II Disposition***
3	Install an independent method of suppression pool cooling.	SAMA would decrease the probability of loss of containment heat removal.	>$2M	[>$1M/Unit x 2] NUREG-1437 cost for independent Containment Spray System is >$1M.	Screened ($)	*As determined in the RAI 3 response, <u>maximum</u> benefit would be to eliminate all loss of containment heat removal sequences (i.e., about 20% of CDF). This would roughly correspond to a $540,000 averted cost-risk using the 3% real discount rate base cost of $2.7M. This is less than a realistic implementation cost for a completely independent suppression pool cooling system for both units, and therefore is still not beneficial.*
5	Install a containment vent large enough to remove ATWS decay heat.	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	>$2M	[$300K/Unit x 2] - Ref. 17, Section A.5.11.1, but installation of hard pipe vent at PB cost >$2 million (Ref. 18)	Screened ($)	*As determined in the RAI 3 response, <u>maximum</u> benefit would be to eliminate all ATWS scenarios (i.e., about 10% of CDF). This would roughly correspond to a $270,000 averted cost-risk using the 3% real discount base cost of $2.7M. This is less than the estimated implementation cost, and therefore is still not beneficial.*

TABLE 8-1 Sensitivity of Phase II Dispositions Using A 3% Real Discount Rate						
Phase II SAMA ID number	**SAMA title**	**Result of potential enhancement**	**Estimated Cost**	**Comment**	**Original Phase II Disposition**	***Revised Phase II Disposition***
23	2.h. Safety Related Condensate Storage Tank	SAMA will improve availability of CST following a Seismic event	>$2M	[>$1M x 2] - Ref. 17, Section A.5.2.4	Screened ($)	*It is judged that a minimal amount of benefit would be obtained from installing a seismically qualified CST. Therefore, the costs of installing such a system are judged to far outweigh the maximum benefit even if the 3% real discount rate is used.*
24	3.c. Improved Vacuum Breakers (redundant valves in each line)	SAMA reduces the probability of a stuck open vacuum breaker.	>$2M	$'s per engineering judgment. ABWR, Section 5.3.3 lists cost as >$100K (per unit). However, this is for a plant not yet built. This is an extensive modification, so cost is estimated at >$1M/unit.	Screened ($)	*The costs of installing such a system are judged to far outweigh the maximum benefit even if the 3% real discount rate is used.*
28	9.i. Dedicated RHR (bunkered) Power Supply		>$2M	[$1.2M x 2] - Ref. 17, Section A.5.9.2	Screened ($)	*As determined in the RAI 3 response, maximum benefit from eliminating all SBO sequences using the mean CDF value is ~ $285K. This corresponds to an averted cost-risk of $381,082 using the 3% real discount rate. Based on estimated implementation costs of >$2M, the SAMA is still not beneficial.*

The remaining Phase 2 SAMAs are dispositioned based on PSA insights or detailed analysis. With the exception of Phase 2 SAMA 6, the PSA insights used to screen the SAMAs are still applicable given the use of the 3% real discount rate and are not investigated further. The SAMAs eliminated based on the results of their detailed analyses are considered to be bounded by Phase 2 SAMA 21.

In the response to RAI 6, Phase 2 SAMA 6 was shown to provide benefit primarily in SBO sequences. The maximum averted cost-risk for this SAMA was estimated using the assumption that the implementation of Phase 2 SAMA 6 would eliminate all cost-risk associated with SBO sequences. This cost-risk was estimated to be $284,000 for the site using a real discount rate (RDR) of 7 percent. Applying the 3% real discount rate to this case yields an averted cost-risk of approximately $370,000. The cost of implementation for this SAMA is given in the response to RAI 6 as $1M/site; thus, this SAMA is not considered to be cost effective even when a real discount rate of 3% is used.

As shown in the response to RAI 3, none of the SAMAs were found to be realistically cost beneficial even when the 95th percentile PSA results were used in lieu of the mean. The baseline case for the 95th percentile and a 7% real discount rate corresponds to a maximum averted cost-risk of $13.5M while the baseline case using the mean and a 3% real discount rate yields a maximum averted cost-risk of $2.7M. The impact of using the 95th percentile PSA results is significantly greater than the impact of applying a 3% real discount rate. The analysis for the response to RAI 3 did, however, apply more realistic cost estimates for SAMA implementation to complete the screening process. In the case of the 3% real discount rate sensitivity, all of the SAMAs are screened using the original estimates for the costs of implementation. Table 8-2 provides a summary of these cases.

Table 8-2: Summary of Sensitivity to the Real Discount Rate (RDR) on the Cost Benefit

Phase 2 SAMA ID	Averted Cost Risk (7% RDR)	Averted Cost Risk (3% RDR)	Cost of Site Implementation	Net Value (using 3% RDR)
1	$8,409	$11,016	$50,000	-$38,984
11	$265,097	$347,277	$1,600,000	-$1,252,723
13	$388	$508	$50,000	-$49,492
18(a)	$93,785	$112,858	$2,000,000	-$1,887,142
18(b)	$174,238	$228,252	$2,000,000	-$1,771,748
21	$350,956	$460,631	$480,000	-$19,369

While the potential exists for the choice of the real discount rate to change the net value of borderline cases from positive to negative or from negative to positive, the impact of these types of changes on the decision making process should be small. Borderline cases require other engineering analyses as the primary decision making tools. In conclusion, the choice of the real discount rate does not significantly impact the Peach Bottom SAMA analysis.

There are other variables in the SAMA analysis that could realistically assume a range of values. These variables include items such as evacuation timing assumptions, population and meteorology data, property values, costs of implementation, and the effectiveness of proposed SAMA modifications. These factors either have a small impact on the results or are accounted for in the method of the analysis.

For example, while the effectiveness of evacuating the relevant population during an accident is difficult to assess, there is little variance in the results based on the values assigned to the evacuation parameters. This sensitivity was performed as part of NUREG/CR-4551.

The estimated costs of implementation are typically below the actual cost of implementation due to additional analysis and labor that were not considered in the conceptual stages of planning. Lower costs of implementation reduce the likelihood that SAMA candidates will be screened because they are "not cost beneficial". Thus, in the SAMA analysis, low estimates for cost of implementation are conservative as they retain SAMAs for more detailed analysis when those candidates could be screened given a more realistic estimate for the cost of implementation. The impact of the values derived for the costs of implementation is judged to be low.

Another variable is the assumed effectiveness of the SAMA enhancement. The method chosen for representing SAMA enhancements in the PSA model is to overestimate the impact of the change. For instance, if a SAMA is being considered that would improve the Containment Heat Removal (CHR) capability of the plant, the enhancement is modeled as 100% effective such that all loss of CHR sequences are mitigated. This results in a greater cost benefit for the SAMA and a greater likelihood that the candidate will be retained. In cases where the results of this coarse method of evaluation do not provide a clear indication of the SAMA's worth, more realistic estimates are taken from similar systems already modeled in the Peach Bottom PSA or from other industry PSAs.

The impact of the variation in the meteorology, population data, economic worth of the surrounding area, and other values is considered to be bounded by the sensitivity analysis performed in RAI 3. Use of the 95^{th}

percentile PSA results magnified the CDF and LERF by over a factor of 6 and the conclusions of the analysis remained the same. The impact of reasonable variations in these variables is considered to be small compared with the result of implementing the 95th percentile CDF and LERF values.

9. ***In NUREG/CR-4551, Volume 3, Sections 6.4 and 7.2, one modification to reduce CDF was identified whereby the impact could be large while still being within the range of reasonable cost. The modification is procedural in nature and deals with reducing the probability of a common-mode dc power failure. This potential candidate was not identified within the 207 SAMAs considered by the analysis, and no indication was given as to whether this modification has already been implemented at Peach Bottom. Please indicate if this modification has been considered, and if so, what the results of the analysis are.***

Response to RAI 9

We have been unable to verify the exact source of the cited reference. However, NUREG/CR-4550, Volume 4 is for Peach Bottom and does indicate the following:

> "There are features whose availabilities should not be allowed to increase significantly or they could increase the core damage frequency considerably. These include common mechanical failure of the control rods, the probability of two or more stuck-open safety relief valves, battery common cause and independent hardware faults, and miscalibration of the low reactor pressure permissive circuitry for low-pressure cooling."

In any event, the Peach Bottom PSA was reviewed to identify those potential contributors that made a significant contribution to core damage frequency or risk. These contributors were seen as candidates for further investigation regarding cost beneficial modifications to reduce risk. These items can be identified by examining the Fussell-Vesely importance of systems, structures, and components (SSCs).

It is acknowledged that there are also SSCs which may have low Fussell-Vesely importance but high Risk Achievement Worth (RAW). These SSCs can be characterized as having adequate performance as assessed in the PSA, but are features whose unavailabilities should not be allowed to increase because the increase could have a significant impact on the risk profile.

This latter category of SSCs (low F-V, high RAW) does not translate into SSCs that show cost benefit for plant modifications when the best estimate risk profile is used. There may be some set of assumptions (i.e., highly risk averse bias) that could be implemented that would make such SSCs potentially cost beneficial justifying modifications, but such methods are not traditionally used in the realistic evaluation used for SAMA.

One of the SSCs that fall into this latter category is the DC power system. This system can be quantitatively characterized using the common cause failure (CCF) of DC buses and batteries. The associated importance measures relative to CDF are as follows:

SSC	RAW	FV
Station Battery CCF (2A, 2B, 3C, 3D)	6,945	4.3E-05
Unit Battery CCF (2A, 2B, 2C, 2D)	ε[1]	ε[1]
125 VDC Bus CCF (2A and 2C)	40,858[2]	5.7E-04[2]

(1) Truncates out at 3E-12/yr
(2) All other combinations of Bus CCF truncated out at 3E-12/yr

Based on past practice of assessing the dominant risk contributors based on contribution to the best estimate risk profile, the DC CCF SSCs were not included in the SAMA list for evaluation.

Common cause DC power failure could prove to be a serious challenge to the safe shutdown of Peach Bottom (and every other LWR) if it occurred. Prevention of common cause DC power failure is generally contingent upon many diverse factors:

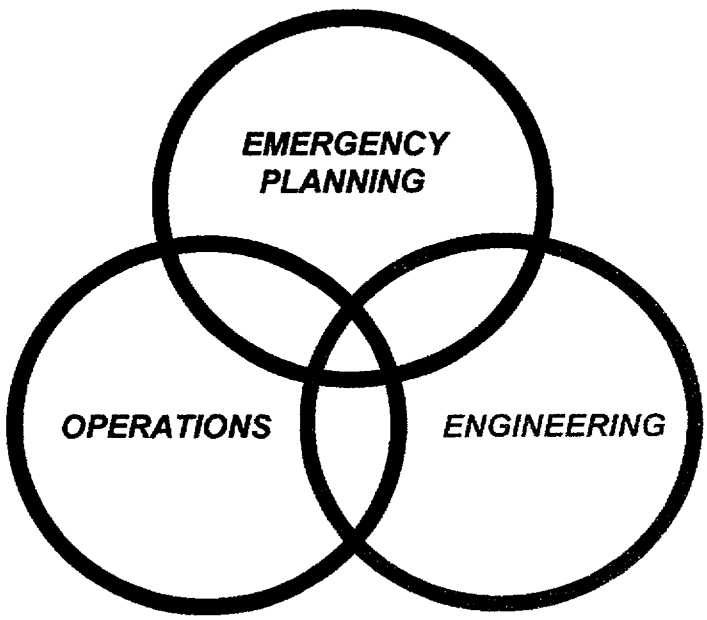
- Maintenance procedures
- Test procedures and frequencies
- Training of maintenance and operation crew
- Clear labeling
- Staggered testing and maintenance
- Diverse power sources
- Avoidance of cross connections among divisions

All of these features are present for Peach Bottom to help protect against a common cause DC power failure.

Summary Response to RAI 9

The intent of the question is assumed to be to identify the potential benefit from reducing battery common cause failures. Battery common cause failures are identified here as a failure mode that would be highly undesirable to increase in probability. This, of course, means that the RAW is large for this basic event. It does not mean that eliminating this failure mode would translate into a decrease in risk. Current methods and procedures at Peach Bottom are structured to minimize the potential for common cause DC failures. The DC system and associated common cause events have a low impact on the baseline CDF and risk. Therefore, justification for a modification is not supported as being cost beneficial.

Evaluation of Peach Bottom Accident Management Insights



Philadelphia Electric Company

October 1998

Peach Bottom Accident Management Program

EVALUATION OF PEACH BOTTOM ACCIDENT MANAGEMENT INSIGHTS WITH REGARD TO BWROG EPG/SAG STRATEGIES

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FOREWORD

This review of the IPE Insights has been performed to support the Accident Management Implementation at Peach Bottom Atomic Power Station (PBAPS).

Note that since the initiation and preparation of the Draft of this study, PECO has drafted and instituted first revisions of the PBAPS Severe Accident Management Procedures (SAMPs) and Technical Support Guidelines (TSGs) (and have revised the EPG based TRIP procedures). Their impact on issues identified in the draft of this report is discussed in the footnotes to Table 3.2-1 and in Section 4, Conclusions. The impact is that all identified insights in this report are now judged appropriately considered and addressed at PBAPS.

Section 1
INTRODUCTION

1.1 BACKGROUND

NEI has published the industry initiative for the Severe Accident Management closure process in NEI 91-04 [1]. This initiative includes a requirement for an evaluation of potential plant specific insights. Specifically, the severe accident management closure process in NEI 91-04 is recommended to include the following step:

> Evaluate industry-developed bases and Owners' Group severe accident management guidance (SAMG) along with the plant IPE, IPEEE and current capabilities, to develop severe accident management guidance for accidents found to be important in your plant as screened with the criteria provided in Section 2.0. Consider other generic and plant-specific information (e.g., NRC and industry studies, PSA results, etc.) as appropriate.

This report addresses this aspect of the industry initiative on closure of Severe Accident Management issues.

As part of the PBAPS PSA process, which includes the Individual Plant Examination (IPE) [3] and the Individual Plant Examination for External Events [6] requested by GL 88-20 [2], PECO has developed a large number of severe accident insights. These insights are documented in the Peach Bottom IPE [3], the NRC staff Evaluation report on the IPE [11], the IPEEE [6], PECO responses to the NRC requests for additional information on the IPEEE [7], and PBAPS Level 1 and 2 PSA Updates [22, 23]. Because the BWROG EPG/SAG development has made significant changes to the approach to severe accident mitigation to incorporate severe accident insights and members of the BWROG EPG/SAG development team also participated in the Peach Bottom IPE, many of the Peach Bottom IPE and IPEEE insights are believed to be addressed by the generic BWROG product and its implementation at Peach Bottom.

1.2 PURPOSE

The purpose of this evaluation is to ensure that insights from the PBAPS Individual Plant Examination (IPE) and Individual Plant Examination of External Events (IPEEE) submittals, and subsequent NRC correspondence regarding these submittals have been factored into the PBAPS specific accident management guidelines.

1.3 SCOPE

The scope of this evaluation is the following: (1) review and identification of all plant specific IPE and IPEEE insights and (2) the disposition of the insights in terms of plant specific actions or potential actions. This review encompasses the following PBAPS documents:

- IPE, Individual Plant Examination Submittal [3]
- NRC SER on the IPE [11]
- IPEEE, Individual Plant Examination for External Events [6]
- PECO responses to NRC request for additional information on the PBAPS IPEEE Submittal [7].
- PBAPS Level 1 and 2 PSA Updates [22, 23].

The PBAPS review has also considered insights from industry results in the following ways:

- NRC documents that have reviewed PSAs and developed insights are incorporated in the EPG/SAG development. [5, 12 - 19]
- NUREG-1150 has been reviewed for insights not covered in the PBAPS IPE/PSA or the generic insights.

- NUREG-1560 (Draft) has been reviewed to determine if any "vulnerabilities" identified by other BWRs apply to PBAPS.

The main focus of the evaluation is to examine the disposition of IPE and IPEEE insights relative to the treatment in the BWROG EPG/SAGs [20] and the representation of these guidelines in the recently revised PBAPS EOP Trip procedures [24] and the newly developed Rev. 0 PBAPS Severe Accident Management Procedures [25].

The BWROG has developed a comprehensive set of generic guidance based on review of NRC and industry insights. This BWROG product consists of Emergency Procedure Guidelines (EPGs) and a separate guideline called Severe Accident Guidelines (SAGs) as discussed below.

The EPGs have retained much of the character and insights in the Rev. 4 EPGs that received a Safety Evaluation Report (SER) from the NRC. Some refinements and improvements in the EPGs have been included. These include:

- ATWS stability response
- Lowering the level from TAF to MSCRWL before which RPV depressurization is required when injection is available.
- Allowing RPV injection at high containment pressures (greater than what was formerly called MPCWLL).

The major modification is in the SAGs where IPE insights and deterministic calculations available from the NRC and industry led to substantial refinement in the action response to potential inadequate core cooling. Specifically, the Containment Flooding Contingency, Contingency 6, from the Rev. 4 EPG was substantially modified to account for additional severe accident information. Because of the extensive revamping of Contingency 6, a separate guideline called the Severe Accident Guideline (SAG) was developed. Therefore, it is in the SAGs that the majority of severe accident insights are addressed.

1.4 GUIDANCE

For the purposes of this evaluation, the procedures and guidance to evaluate the insights consist of the following:

- Revision 1 (1997) Generic EPG/SAGs developed by the BWROG [20]
- PBAPS EOP TRIP, SAMP, Special Event, and related procedures
- Other guidance deemed useful for the PBAPS Emergency Response Organization (ERO) to prevent or mitigate accidents

In addition, the systematic process for evaluating accident management capabilities developed in NEI 92-01 [21] is referenced to ensure that the guidance provided there is consistent with the evaluation performed for PBAPS. See the discussion of the NEI 92-01 interface discussed in Section 2.

Section 2
PROCESS

This section briefly describes the process used in identifying insights and assessing their resolution following implementation of the BWROG EPG/SAGs.

2.1 INTERFACE WITH NEI 92-01

NEI 92-01, A Process for Evaluating Accident Management Capabilities, [21] provides the industry with a process for performing a systematic, structured evaluation of existing and potential accident management capabilities.

As noted in Section 1, NEI 91-04 has established the elements of the industry's severe accident closure plan for the industry. The objective addressed in this analysis is the evaluation of the IPE and IPEEE insights. The NEI 92-01 process is used to assist in that process.

This PBAPS report addresses the following aspects of the NEI 92-01 proposed process:

- Establishing a systematic process for identifying and treating enhancements
- Identification of enhancements or options per Section 4 of NEI 92-01
- Interface of the enhancements with the accident sequence or core melt progression to determine its potential usefulness per Section 6.1 and 6.2 of NEI 92-01
- Integration of options where appropriate

Areas which are not treated in this report but which are covered by NEI 92-01 are the following:

- Ensuring the enhancements are feasible
- Selection of enhancements to be implemented
- Planning for implementation
- Organization issues

The approach chosen for resolution of the Severe Accident Closure objectives identified in NEI 91-04 includes the examination of the IPE and IPEEE insights for PBAPS. The PBAPS assessment of potential enhancements was performed as part of the IPE and IPEEE process and that assessment is not repeated here. Both their insights and accident sequence structures were used to identify the options for enhancement and their interface with accident sequences. The IPE and IPEEE reviews also identified by functional category where options for improvement could be made. Therefore, these documents provide a comprehensive summary of the plant specific enhancements that are useful to consider in addition to the generic accident management insights identified by the BWROG as part of the EPG/SAG BWROG product development.

These enhancement options are then examined in this report to determine whether the BWROG EPG/SAG product and the associated plant specific procedures and training or hardware modifications have adequately addressed the insight. For each PBAPS enhancement, the following areas of disposition are considered:

- Additional procedure or guidance
- Additional training
- Alternative equipment
- Additional instrumentation

The appropriate category of disposition is then identified for each insight or potential enhancement.

In summary, NEI 92-01 has defined a systematic approach to the assessment of accident management insights and their incorporation into the plant specific accident management program. The PBAPS development and review of insights from the IPE, IPEEE, and generic sources mirrors this process. The results of which are included in this report for consideration and possible implementation by the PBAPS Accident Management Program Team.

The following tasks outline the specific assessments performed.

2.2 ACCIDENT PREVENTION INSIGHTS

Those insights identified in the documents cited in Section 1.3 that could be implemented to improve the prevention of accidents are examined to determine if these have been incorporated in the generic EPG/SAGs and the PECO planned implementation at PBAPS, or to identify a disposition.

2.3 ACCIDENT MITIGATION INSIGHTS

Those insights identified in the documents noted in Section 1 that could be implemented to improve the mitigation of severe accidents are examined.

The review of severe accident insights has been performed based on the PBAPS plant specific PSA and these insights are correlated with current plant design and procedures and the proposed EPG/SAG changes (BWROG Document). [20]

2.4 EVALUATION PROCESS

The evaluation process is straightforward in that it:

- takes the identified insights from the referenced documents,
- identifies whether the insight differs from that transmitted to the NRC in the PBAPS IPE or IPEEE Submittals,
- compares the insight to the available guidance in the recently revised PBAPS TRIP and newly developed Rev. 0 Severe Accident Management procedures [24, 25], derived from the BWROG product [20],
- assesses whether the insight is adequately addressed in the PBAPS EOP TRIP and SAMP procedures (or, alternatively as appropriate, in other existing plant procedures, or by completed or proposed training, procedural, and/or hardware enhancements),
- identifies whether additional action is required to close out treatment of the insight.

Section 3
IDENTIFICATION OF PLANT SPECIFIC INSIGHTS

This section summarizes the PBAPS specific insights that have been identified by PECO as part of the severe accident closure process. The process examines the insights from the following sources:

Section	Insights Description/Source
3.1	Level 1 IPE and PSA (Accident Prevention)
3.2	Level 2 IPE and PSA (Accident Mitigation)
3.3	IPEEE (External Events)
3.4	Generic Insights

3.1 LEVEL 1 IPE AND PSA INSIGHTS

In 1988, the NRC issued Generic Letter 88-20 [2] requiring each utility to perform an Individual Plant Examination (IPE) for severe accident vulnerabilities. GL 88-20 stated the following objectives that the NRC expected to be accomplished by the performance of an IPE:

- To develop an appreciation for severe accident behavior at PBAPS
- To understand the most likely severe accident sequences that could occur at PBAPS
- To gain more quantitative understanding of the overall probabilities of core damage and fission product releases
- If necessary, to reduce the overall probabilities of core damage and fission product releases at PBAPS by modifying, where appropriate, hardware, procedures, or training that would help prevent or mitigate severe accidents.

In order to satisfy the requirements of GL 88-20, PECO elected to perform an IPE for PBAPS by utilizing a Probabilistic Safety Assessment approach. As requested in GL 88-20, the PBAPS IPE consists of both a Level 1 and a Level 2 PSA. The PBAPS Level 1 PSA is an integrated analysis of plant and system responses to a wide spectrum of internal events such as reactor scrams, loss of off-site power, loss-of-coolant accidents, and other special initiators. The Level 2 PSA considers core damage timing and subsequent containment challenges to quantitatively assess the potential for significant release of radioactivity.

Following the submittal of the PBAPS IPE to the NRC in August 1992, PECO received NRC acceptance and the staff evaluation report of the Peach Bottom IPE Submittal. [11] No request for additional information regarding the PBAPS IPE was issued by the NRC.

Following the submittal of the IPE and consistent with the PECO philosophy of maintaining an up to date, usable PSA, the PBAPS Level 1 IPE models were updated in 1997. [22] Enhancements, modifications, and corrections to the models in support of the 1997 PSA Update resulted in Level 1 and Level 2 results and conclusions very similar to the 1992 IPE results; as such, no additional insights beyond those reported in the IPE Submittal are provided by the 1997 Update.

A substantial number of risk insights have been identified and documented by the above studies. The insights aimed at preventing core damage, i.e., Level 1 insights, and their status relative to the EPGs/SAGs are summarized in Table 3.1-1. Table 3.1-1 is constructed to provide the following information:

- Insights that can be derived from the examination of the IPE submittal or the latest PBAPS PSA Update
- Document used to identify each insight

- Page/location in the source document
- Determination as to whether the insight is the same as that appearing in the IPE Submittal to the NRC
- Assessment of the treatment and disposition of the insight at PBAPS. This assessment is made based on the Revision 1 of the BWROG EPG/SAG [20], and the recent first drafts of the EPG/SAG-based PBAPS EOP TRIP and SAMP procedures. [24, 25] Also considered are PBAPS Special Event procedures, training and hardware modifications.
- Finally, assessment of whether additional action is necessary to adequately incorporate the insight into plant specific PBAPS guidance, procedures, training or hardware enhancements. An "N" signifies "No"; that is, no additional action is judged necessary to address the insight. A "Y" signifies "Yes"; that is, additional action is necessary or prudent to fully address the insight for PBAPS.

Refer to Section 3.2 of this report for insights aimed at successfully mitigating postulated severe accidents, i.e., Level 2 insights. In addition, Section 3.3 provides the insights developed in the Individual Plant Examination for External Events (IPEEE).

3.2 LEVEL 2 IPE AND PSA INSIGHTS

The containment performance evaluation requested as part of the NRC's Severe Accident Policy Statement resulted in the performance of a Level 2 PSA analysis by PECO This Level 2 PSA was submitted as part of the GL 88-20 (IPE) response. The Level 2 analysis is very detailed and includes the probabilistic quantification of Containment Event Trees (CETs) including phenomenological impacts. There are detailed thermal hydraulic calculations using the MAAP computer code to model the core/debris state along with the RPV and containment conditions. The MAAP code also tracks the fission product distribution during the accident progression.

Based on this detailed PBAPS analysis, there have been a substantial number of insights developed regarding potential actions that could be taken to enhance containment performance. These insights are aimed at increasing the safety of the public by providing directions on successfully mitigating postulated severe accidents. The BWROG EPG/SAG development team included members who were involved in the development of the PBAPS IPE insights. Therefore, the BWROG benefited from the PBAPS IPE insights and as might be expected a large number of the insights have been addressed by the new EPG/SAGs. There were, however, in the BWROG EPG/SAG product [20], a small number of insights that were judged to be more appropriately treated in training rather than proceduralized and further a number of insights that conflicted among themselves. Therefore, some insights from the IPE are not explicitly incorporated in the generic EPG/SAG products from the BWROG.

Like the Level 1 IPE models, the Level 2 models were Updated in 1997 [23] and results do not provide additional insights beyond those reported in the IPE Submittal.

The insights and their status relative to the EPG/SAG are provided in Table 3.2-1. Table 3.2-1 is constructed to provide information similar to that described above for Table 3.1-1.

3.3 EXTERNAL EVENTS

In June 1991, the NRC issued Supplement 4, Individual Plant Examination of External Events (IPEEE), to Generic Letter 88-20 requesting each utility to investigate potential severe accident vulnerabilities posed by external hazards. As requested in Supplement 4 of GL 88-20, the PBAPS IPEEE is an examination of the following hazards:

1. Seismic events
2. Fires
3. External Floods
4. High Winds and Tornadoes
5. Transportation and Nearby Facility Hazards

6. Other Plant-Unique Hazards

External hazard risk insights and their status relative to the EPGs/SAGs are summarized in Table 3.3-1. These insights turned out to be primarily related to Level 1 actions for prevention of core damage including equipment restoration. The PBAPS IPEEE provided no additional Level 2 insights (i.e., to improve containment performance under severe accident conditions) other than those already identified in the IPE (see Section 3.2 of this report).

3.4 GENERIC INSIGHTS

The review of a large number of generic documents [5, 12-19] was performed by the BWROG who incorporated the relevant insights into the BWROG EPGs and SAGs. As noted in Section 3.2, the BWROG EPG/SAG development team included members who also developed the insights for the PBAPS IPE. These generic insights were reviewed for applicability on a plant specific basis. No additional insights beyond those addressed by the BWROG EPG/SAG development were identified. Therefore, these documents were not investigated further.

The review of NUREG-1150 and the supporting documents (e.g., NUREG/CR-4550 and 4551) identified a number of preventive measures that were already included in the EPGs and also several containment performance issues that had previously been addressed generically by the BWROG in the EPG/SAG development.

The NUREG-1560 (DRAFT) from the NRC was also reviewed for additional NRC perspectives in the BWR IPEs. [26] The NRC has compiled areas referred to by individual licensees as "vulnerabilities" and also plant improvements. These two categories of insights have been provided in NUREG-1560 (DRAFT). Table 3.4-1 summarizes the so-called vulnerabilities and the PBAPS status relative to these vulnerabilities. As noted, no additional PECO actions are required.

Table 3.4-2 summarizes other areas of improvements identified by licensees. These other improvements were also considered by PECO in the review for accident management insights and no additional action is deemed appropriate. A number of individual issues were identified, however, these were found to be either already addressed by the BWROG or were plant specific issues that did not apply to PBAPS.

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Loss of Offsite Power (LOOP) Procedure SE-11</u> The Peach Bottom IPE Loss of Offsite Power event tree models represent the situation in which both units lose offsite power. This represents the more significant challenge, requiring considerations of electrical loads and operator actions necessary to safety shutdown U3 in addition to U2. The shared electrical configuration at Peach Bottom limits operator actions and some equipment availability at both PBAPS units. The diesel generator load capacity (one RHR pump per diesel) and the electrical distribution can lead to core damage scenarios with available diesel generators. The issues above resulted in station blackout scenarios contributing about 50% to the core damage frequency in the Peach Bottom NUREG/CR-4550 analyses. The Peach Bottom IPE recognizes these issues and divides LOOP scenarios into SBOs (i.e., failure of all diesels) and scenarios with various combinations of diesels available and failed, requiring enhancements to LOOP procedure SE-11 to allow credit for cross-tie of power sources.	a) IPE Submittal b) IPE SER	a) 3.1-43, 6-1, 8-8 b) 27, 36	Y	Enhancements were made to the LOOP procedure SE-11 to cross-tie emergency buses and recognize inter-unit interactions. This proactive enhancement involves a significant change in the response to LOOP events. Entry conditions are not limited to Station Blackout but involve all permutations of diesel failures (none, 1, 2, 3, or 4) coincident with a LOOP. Equipment prioritization considers the requirements of both Peach Bottom units in achieving a safe and controlled shutdown following a LOOP. Training was implemented to instruct operators in the procedural enhancements. No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Alternate Water Supply for Drywell Spray/Vessel Injection</u> In enclosure 2 to Supplement 1 of Generic Letter 88-20, the NRC identified certain containment performance improvements that would reduce the vulnerability of Mark I containments to severe accident challenges. One such improvement is the ability to connect a backup or alternate supply of water, one independent of normal and emergency AC, that could be delivered to the reactor vessel and/or drywell via the RHR system. Such an alternate source of water to the reactor vessel reduces the likelihood of core melt as well as provides significant accident management capability (e.g., debris cooling, fission product scrubbing). PBAPS has the capability to inject both HPSW and the fire water system into the reactor vessel or containment via the RHR system. Although HPSW is powered by onsite emergency AC, its use is enhanced by the electrical cross-tie directions of the enhanced SE-11 procedure. The fire water source is independent of AC power.	IPE Submittal	6-2, 6-5	Y	The hardware and procedures already exist at PBAPS for alternate injection. Procedure T-245 is used to align the RHR/HPSW cross-tie for RPV injection, and T-205 is used to direct alignment for containment sprays. Procedure T-243 is used to align the fire water system for RPV injection. Both methods, in addition to others, are identified in the PBAPS EOP TRIP and SAMP procedures. No modifications to the EPGs/ SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Enhance RPV Depressurization</u> Another GL 88-20, Supplement 1 identified Mark I containment performance improvement is enhanced depressurization capability to reduce the likelihood of loss of coolant injection scenarios. At Peach Bottom all eleven SRVs are provided with two redundant 125 VDC power supplies to the solenoid of each valve. The containment isolation valves that provide long term nitrogen supply to the ADS valves are powered from emergency buses. In addition, the normal nitrogen supply to the ADS valves is backed by bottles and an outside connection for long term nitrogen supply.	IPE Submittal	6-2, 6-4	Y	Both the EOP TRIP and SAMP procedures recognize the benefit of RPV depressurization and direct appropriate steps in addition to referencing the procedures for system backups: • SO 16A.7.A, Backup N2 to ADS • GP-8E, N2 Isolation Bypass • T-261, CAD Tank Backup to N2 In addition, the enhanced SE-11 LOOP procedure recognizes the need to provide emergency power to the ADS valves. No modifications to EPG/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
Torus Hard Pipe Vent The containment hard pipe vent was installed in Units 2 and 3 after the IPE Submittal, but is incorporated into the analysis. The analysis assumes that containment venting for containment heat removal purposes is performed using the 16" torus hard pipe vent. Use of the hard pipe vent reduces the likelihood of subsequent coolant injection failure caused by the release of large amounts of steam into the reactor building that would otherwise occur if the vent paths through SGTS were employed. The ability to perform containment venting, and particularly the hardpipe vent feature, significantly reduce the CDF contribution of scenarios involving loss of containment heat removal.	a) IPE Submittal b) IPE SER	a) 3.1-31, 6-3, 6-4 b) 63	Y	The EOP TRIP and SAMP procedures specifically address venting. This issue was addressed by the BWROG and prioritization of venting from the torus is considered desirable. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
Turbine Trip ATWS The status of PCS and FW are very important to the accident sequence development of ATWS scenarios, in which the main condenser is initially available as a heat sink. The discharge of a substantial fraction of steam to the main condenser (i.e., as opposed to into the primary containment) affords the operator more time to perform actions (e.g., SLC injection, lower water level, depressurize RPV) than if the main condenser was unavailable, resulting in lower human error probabilities.	IPE Submittal	3.1-80, 3.3-26 to 3.3-36	Y	BWROG EPC Issue 98-07 addresses this issue. The bypass of the MSIV isolation was moved upward in the flowchart, rendering it more important. PBAPS implementation has followed the BWROG recommendation in placement of this step. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Operator Actions During ATWS</u> Operator actions play a dominant role in the progression ATWS sequences. There are four basic goals of the main control room (MCR) crew response to an ATWS: (1) reactivity control, (2) reactor vessel water level control, (3) reactor vessel pressure control, and (4) pressure suppression pool temperature control. The operator is expected not only to attempt to insert control rods but to be very aware of the limitations of the Standby Liquid Control System (SLCS) and how void effects can be used to shutdown the reactor. Normal depressurization is called for at a torus temperature of 110°F. More rapid depressurization is allowed as the torus temperature reaches the torus heat capacity temperature limit (HCTL) of approximately 180°F. The operator is directed to emergency blowdown to maintain torus temperature below the HCTL curve. The system time windows used in deriving HEPs for the different operator responses are based on the time to reach specific suppression pool temperatures as called out by the TRIP procedures.	IPE Submittal	3.1-90 to 3.1-91	Y	Operator actions during ATWS scenarios are clearly directed in the EOP TRIP procedures and receive attention in training. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Refill CST</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to augment and delay depletion of the tank. This would reduce the risk of core damage during events such as extended station blackouts or LOCAs which render the suppression pool unavailable as an injection source due to heat up.	IPE Submittal	6-4	Y	Capability exists to transfer water from the RWST or other unit's CST to the affected unit's CST. This is proceduralized in the Loss of Offsite Power Procedure SE-11. It has also been added to SAMP-1, Sheet 1 at RPC/F1.1.	N
<u>Maintain ECCS Suction on CST</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to maintain suction on the CST as long as possible to avoid pump failure as a result of high suppression pool temperature. Manual switchover between the CST and suppression pool suctions for HPCI and RCIC during a station blackout is not modeled in the IPE. Successful manual switchover, and failure to switch back to the CST when the suppression pool temperature increases beyond 220°F would result in failure of HPCI and RCIC. Similarly failing to switch over to the suppression pool initially results in a relatively early loss of HPCI and RCIC due to switch over occuring when the pool is hot.	IPE Submittal	6-4	Y	Swap to/from CST source is procedurally directed.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source: Document	Source: Page	Source: Same as IPE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
Shed Non-Essential DC Loads Generic letter 88-20, Supplement 2 identified this accident management strategy to shed non-essental DC loads to conserve battery power during station blackout scenarios for operation of essential equipment as long as possible.	IPE Submittal	6-4	Y	LOOP procedure SE-11, Attachments P, T and Y provide guidance on load management, shedding of DC loads and restoration of DC loads. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
Use Portable Chargers Generic letter 88-20, Supplement 2 identified this accident management strategy to recharge station batteries during a station blackout to prolong the available of DC Power.	IPE Submittal	6-4	Y	LOOP procedure SE-11 directs cross-tie electrical buses, allowing chargers to be supplied from other divisions. The procedure specifically directs supplying power to all battery chargers (if possible). Procedural and hardware enhancements maybe pursued to allow use of portable battery chargers, but is not crucial considering the extensive cross-tie capability provided by SE-11. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source: Document	Source: Page	Source: Same as IPE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
<u>Early Detection and Mitigation of ISLOCA</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to limit the effects of ISLOCA accidents by early detection and isolation.	IPE Submittal	6-4	Y	Enhanced procedural guidance or training regarding how to detect and pinpoint the location of an ISLOCA and isolate it may improve the likelihood of early isolation. However, given the low risk contribution for ISLOCAs (< 1% of CDF and ~3% of LERF) at PBAPS, such enhancements are not crucial. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>CRD Injection</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to supply an additional method of level restoration by using a non-safety system.	IPE Submittal	6-4	Y	Maximization of CRD is covered in the existing EOPs which appropriately refer to T-246 for detailed directions. In addition, for LOOP events, procedure SE-11, Attachment W provides guidance regarding alignment of cooling to maintain CRD availability. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Condensate Pumps for Injection</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to provide an additional option for coolant injection when other systems are unavailable or inadequate.	IPE Submittal	6-5	Y	The use of condensate is covered in existing EOPs and in training. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>Align EDG to CRD</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to provide power to an additional injection source during loss of power events.	IPE Submittal	6-5	Y	CRD pumps at PBAPS are normally fed from diesel-backed emergency 4 kV buses. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>Guard Against SLC Dilution</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to control vessel injection to prevent boron loss or dilution following SLC injection.	IPE Submittal	6-5	Y	SLCS initiation and existing procedures guard against dilution (RWCU isolation and overfill prevention).	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Additional Supply of Borated Water</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to ensure long term supply of borated water.	IPE Submittal	6-5	Y	Although this is primarily a PWR concern, PBAPS does have the capability to inject backup supplies of boric acid and borax into the RPV using the CRD pumps. The steps to perform this activity are provided in procedure T-211, appropriately referenced in the EOP TRIP procedures. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>Re-open MSIVs</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to regain the main condenser as a heat sink by re-opening the MSIVs. This strategy requires that condenser vacuum be maintained or re-established and that circulating water be available.	IPE Submittal	6-5	Y	Existing EOPs direct this including bypass of low level interlocks as necessary.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Bypassing RCIC Turbine Exhaust Pressure</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to enable continued RCIC operation beyond the point at which it would normally trip.	IPE Submittal	6-5	Y	Peach Bottom does not have procedures in-place for bypassing the exhaust trip. Bypassing the protective trip or changing the setting could be detrimental and result in the need for constant operator vigilance and dependence on the adequacy of existing instrumentation. In any event, the RCIC turbine exhaust pressure trip is sufficiently high (50 psig) such that it will not be reached for most accident types until many hours (10 - 20). As such, the benefit of such a procedure in reducing plant risk is minimal. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.1-1

IPE/PSA INSIGHTS (LEVEL 1)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May Be Req'd (Y/N)
	Document	Page	Same as IPE (Y/N)		
<u>Bypassing of Trips for Diesels</u> Generic letter 88-20, Supplement 2 identified this accident management strategy to enable continued emergency diesel generator operation beyond the point where they would normally trip. This strategy is accomplished by bypassing certain protective trips or changing their trip setpoints. EDGs are typically designed with automatic bypass of some protective trips during emergency start. Examples of the types of trips typically bypassed during emergency starts are: high jacket water temperature, high vibration, low tubo rcharger lube oil pressure, main bearing high temperautre, and connecting rod bearing high temperature. Other trips which are found to be automatically bypassed in some plants are low lube oil pressure, high crankcase pressure, and generator-differential. If automatic bypass of any of these trips is not presently part of the system design, they may be candidates for manual bypass.	IPE Submittal	6-5	Y	Many trips are automatically bypassed on "LOCA start" of diesel. In addition, SE-11 covers troubleshooting of diesel trips and provides guidance on resetting trips and restarting EDGs. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
The large LOCA accident class is the largest single contributor, representing about 55% of the H/E release category.	IPE Submittal	4.6-35	Y	SAMP-1 (SH 2,3, 4 and 5) have incorporated EPG/SAG actions to use external water sources for mitigation. This will provide the best potential mitigation.	N
It is also useful to examine the variation in the contributors to core damage and large releases. This comparison graphically shows that the sequences that dominate the high release. For example, ATWS sequences comprise a significant percentage of the core damage frequency, yet result in a small percentage of high releases. On the hand, Class IA sequences represent a significant percentage of both core damage frequency and high magnitude release frequency. In both these cases, the tendency to result in a certain release magnitude is a function of the accident sequence characteristics and not simply percentage of core damage frequency.	IPE Submittal	4.6-38	Y	SAMP-1 and SAMP-2 incorporated EPG/SAG guidance to address different causes of severe accidents.	N
Containment isolation failure is treated conservatively in the assignment of Radionuclide Release state, sequences are assigned a high release in the case of IS failure even though: • The failures could be relatively small • The failures could be from the torus air space • The failures could be into closed or filtered systems (e.g., SGTS)	IPE Submittal	4.7-29	Y	Containment isolation failure found not to be dominant despite the conservative treatment.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>Containment Flooding</u> Events during which the containment flood contingency is successfully implemented and completed, are found to have the possibility of direct releases from the RPV to the condenser and from the DW through the DW vent. For Peach Bottom, the DW vent consists of ductwork that is assumed to fail in the Reactor Building. Because of the nature of this accident scenario, there are no reference plant calculations publicly available. Therefore, conservative estimates are used to characterize the release categories as follows: • Successful containment flood, but ineffective reactor building, condenser, or turbine building effectiveness results in a high (H) release. • With effective mitigation by secondary buildings, the release is classified as moderate.	IPE Submittal	4.7-30	Y	EPG/SAG development has specifically addressed the question of RPV venting. RPV venting has been either precluded by procedure or significantly delayed. This results in a substantial improvement in mitigation capability by minimizing radionuclide release potential without compromising recovery of core or debris cooling. SAMP-1 and SAMP-2 incorporate these guidance items for the EPG/SAG.	N[1]
Use of the hard pipe vent results in bypassing the reactor building. Therefore, the reactor building node is not considered in sequences where CV = success.	IPE Submittal	4.7-31	Y	This issue was addressed by the BWROG and venting from the torus is considered desirable to protect containment. The suppression pool scrubbing is depended upon to provide substantial mitigation of radionuclide release.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>High Pressure Sequences</u> Transient initiated sequences with subsequent unavailability of high pressure injection and initial failure to depressurize (human error) is the dominant core damage sequences. (Level 1)	IPE Submittal	3.4-2	Y	BWROG EPG/SAG recognizes the importance of depressurization and it continues to be emphasized as a critical operator action in the procedures and in training.	N
<u>In-vessel Recovery</u> The PBAPS EOPs direct the restoration of adequate core cooling even during degraded core states. Only minimal credit has been given in the analysis for this in-vessel recovery. This may be a conservatism in the analysis.	Level 2 IPE Tier 2 Document	C.4-23	Y	The EPG/SAGs emphasize the restoration of RPV injection for adequate core cooling. This has been elevated in importance even above preservation of containment integrity when MPCWLL is challenged. PBAPS EOP Update and SAMP-1 and SAMP-2 implement the EPG/SAG revised decision process.	N
<u>Depressurization</u> The ability to depressurize the RPV during core melt progression, i.e., prior to RPV breach by molten debris can be a major influence on the determination of the accident sequence timing, phenomena that occur, and the challenge applied to the containment. These effects are reflected in the Level 2 model in four principal ways:	Level 2 IPE Tier 2 Document	4.6-2	Y	BWROG EPG/SAG recognize the importance of depressurization and it continues to be emphasized as a critical operator action in the procedures and in training.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source: Document	Source: Page	Source: Same As IPE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Depressurization</u> (cont'd) • The sequence can be completely altered by modifying the conditional probability of subsequent event tree nodes dependent on the pressure status of the RPV.				Symptoms for emergency depressurization were considered adequately treated by the BWROG in the transfer to SAGs. • RC/L (MSCRWL now specified as level above which ED is required instead of TAF when injection is available) • Contingency 1 (MSCRWL now specified as level above which ED is required instead of TAF when injection is available) • Contingency 3: Steam Cooling remains the same (delayed until MZIRWL) The PBAPS TRIPS use TAF as the water level decision point. RC/L directs the operator to Contingencies 1 and 3, which direct the action at TAF.	N
• Radionuclide release end states may be altered as a result of the status of RPV depressurization.				If RPV breach from high pressure occurs into an open containment, the radionuclide releases are substantially higher than the case with the RPV depressurized at the time of failure.	N
• The challenge to containment can cause actions or failures not otherwise implemented.					N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>RPV Injection</u> Water injection to the RPV versus the containment sprays has a number of beneficial features which include: • Cooling residual core material in the bottom head • Cooling fuel rods that remain intact in the core region • Cooling by steaming the fission products that are plated out on RPV internal surfaces (dryer/separator). The ability to provide all of these cooling benefits varies with the water source, i.e., the injection source and its flow rate. The following RPV injection sources are considered viable and have the following benefits or disadvantages: Core Spray: This appears to be the most desirable[1] injection source for severe accident mitigation and minimizing radionuclide releases. The core spray system has a relatively high flow rate and produces a spray pattern that is most conducive to cooling material in the RPV given that the RPV bottom head has been breached during core melt progression.	Level 2 IPE Tier 2 Document	4.6-4	Y	Core Spray prioritization has been included in the SAGs as a principal response system for the following SAG "legs" and their conditions: • RC/F-3: RPV Level above BAF • RC/F-4: RPV injection greater than MDRIR • RC/F-1: RPV Vessel Breached (when containment water level is restored above TAF) For other legs, CS is still considered in the context of water sources that can support injection, but it is not prioritized. Reference the following steps: SAG • RC/F-3, 4[(1)] • RC/F-1.3	N

[1] Note that conflicting conclusions may be reached using current T&H codes for sequences in which there is a failure to scram and the RPV is intact.

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
RPV Injection (cont'd) Water will also run out the bottom head of the vessel through the breach and fall on the debris on the drywell floor. This results in the potential to also cool the debris on the drywell floor. LPCI: This is the next most desirable injection source. It has all the advantages cited for Core Spray except that it is injected in the recirculation lines and results in the possibility of being short circuited past the core region and directly out the bottom head breach. This has the possibility of allowing revaporization in the extremely long term as one of its disadvantages. This could be most important in containment flood scenarios when RPV venting is directed by the EOPs where the revaporization source term may escape directly through the RPV vent. HPSW: This has identical attributes to LPCI except a continuous supply of cool water is available; the LPCI recirculates water from the suppression pool. CRD: This water source is desirable but is of limited flow rate. In addition, after RPV breach flow path may not allow delivery to the RPV or to the drywell. This system is not considered here as an effective mitigating system for severe accidents that have progressed outside the RPV.	Level 2 IPE Tier 2 Document			Use of external water sources is heavily emphasized in the SAGs SAMP-1 and SAMP-2 implement this philosophy.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>RPV Injection Versus Drywell Sprays</u> Injection to the RPV has all the advantages that were discussed above in the RPV injection discussion. Drywell sprays have many of the advantages of the RPV injection method including maintaining low drywell temperatures; however, the use of drywell sprays would be marginally effective in cooling debris that was retained in the vessel. In addition, if the operators were able to enter into containment flooding then RPV venting would be directed and the use of drywell sprays during RPV venting may also have a minimal impact on the release directly from the RPV.	Level 2 IPE Tier 2 Document	4.6-5	Y	The DW spray priority is explicitly set in the steps relative to RPV injection. This assures the proper allocation of water resources. • DW Sprays are prioritized • Restrictions on DW Spray operation are relaxed compared to Rev. 4 EPGs. • Drywell spray initiation is called for in the Primary Containment Radiation Guideline of the SAGs The SAG has been structured to attempt drywell spray initiation under severe accident conditions. This includes the new symptom of high radiation which is useful in ensuring drywell spray initiation prior to RPV breach. Reference the following steps: SAG - RC/F-1 - RC/F-5 - RC/F-2 - RC/F-6 - RC/F-3 - PC/H - RC/F-4 - PC/R	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
RPV Injection Versus Drywell Sprays (cont'd)				Drywell sprays also are recognized to provide a number of important mitigation functions. Therefore, drywell sprays are also given special treatment when allocating limited resources for water injection. • RC/F-1: RPV Vessel Breached -- Drywell spray is not restricted by RPV injection and therefore drywell spray has been chosen as the principal water injection source • RC/F-2: RPV Level above TAF -- Maintenance of RPV level above TAF is considered the principal goal, therefore drywell sprays are only used if directed and they do not interfere with maintaining RPV level above TAF • RC/F-3: RPV Level above BAF -- Maintenance of RPV level above BAF is considered the principal goal, therefore drywell sprays are only used if directed and they do not interfere with maintaining RPV level above BAF	

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
RPV Injection Versus Drywell Sprays (cont'd)				• RC/F-4: RPV Injection above MDRIR -- Maintenance of RPV injection flow above MDRIR is considered the principal goal, therefore drywell sprays are only used if directed and they do not interfere with maintaining RPV injection flow above MDRIR • RC/F-5: RPV Injection less than MDRIR, but PSP maintained -- Drywell spray is specified as number 1 priority. This hopefully increases the pressure suppression capability of the containment. • RC/F-6: RPV Injection less than MDRIR, but PSP exceeded -- Drywell spray is specified as highly desirable as long as RPV injection is not reduced.	

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
Debris Cooling Coolant injection to the drywell via either the RPV or the drywell sprays has the benefit of providing debris cooling. This cooling will have the following beneficial effects: • Limit temperature increase in the drywell during the core melt progression • Limit the non-condensable gas generation in the containment and, thereby, prevent reaching the critical containment failure pressure and temperature.	Level 2 IPE Tier 2 Document	4.6-5 to 4.6-6	Y	See discussion under RPV injection and under DW sprays	N
Containment Wetwell Venting or Wetwell Breach With Continued Injection Containment venting provides a useful method of containment pressure control and containment heat removal. If continued coolant injection to the containment can be maintained despite the core melt progression outside the vessel and despite the containment venting procedure, then radionuclide releases can be minimized. Much of this discussion also applies to situations in which the wetwell airspace may fail.	Level 2 IPE Tier 2 Document	4.6-8 to 4.6-9	Y	Additional flexibility in the timing of containment venting has been included in EPG/SAG and in turn in the SAMP-1 and SAMP-2. TSG 3.3 contains specific guidance on the timing of venting.	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source: Document	Source: Page	Source: Same As IPE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Containment Wetwell Venting or Wetwell Breach With Continued Injection (cont'd) Different cases of containment venting are found to result in substantially different estimates of the radionuclide release. For example, consider the two cases below: Case 1: Maintain Injection to RPV or Containment YES; Wetwell Vented YES; Suppression Pool Bypass NO Case 2: Maintain Injection to RPV or Containment YES; Wetwell Vented YES; Suppression Pool Bypass YES The results of MAAP calculations indicate that: 1. Case 1: The radionuclide releases are very low (LL) for the case in which water injection, wetwell venting, and no suppression pool bypass are present 2. Case 2: Releases are approximately 10-100 times larger for the case in which suppression pool bypass is present.					

Cases referenced in the insight above:

	Maintain Injection to RPV or Containment	Wetwell Vented	Suppression Pool Bypass
Case 1	YES	YES	NO
Case 2	YES	YES	YES

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
Containment Wetwell Venting or Wetwell Breach With Continued Injection (cont'd) The purpose of venting is to avoid containment over-pressurization and protect the containment structural integrity. Functionally, this can be accomplished by using the system designed for containment venting or combustible gas control. Additionally, the containment can be successfully vented through a breach in the structure. The impact of venting on a potential environmental source term is dependent primarily on two factors: 1. Timing for establishing the vent pathway; and 2. The suppression pool effectiveness, i.e., the availability of a pathway that routes the radionuclides through the suppression pool.					
Vent: Timing of Radionuclide Release The timing of containment venting can influence the radionuclide release by: • Releasing material early in an accident scenario • Minimizing blowdown flow from the containment that would otherwise occur from an uncontrolled containment rupture These effects on radionuclide release magnitude are considered small compared to the effect assigned to timing of release.	Level 2 IPE Tier 2 Document	4.6-9	N/A	Address vent pathway prioritization and timing in training, TSGs, or additional guidance documents.	Y(2), (3), (4)

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>Vent: Suppression Pool Scrubbing of Element</u> Suppression pool water temperature (i.e., degree of subcooling) may affect the characteristic of the pool to retain aerosols during the vent. It is postulated that as the bulk temperature of the pool approaches saturation temperature, the effective DF of the pool decreases. In fact, the surrogate MAAP calculations indicate that upon reaching saturation temperature, the pool DF becomes unity (i.e., all aerosol radionuclides pass through the pool).	Level 2 IPE Tier 2 Document	4.6-9	N/A	Flexibility to vent at low pressures provided by SAGs. Decision to vent deferred to training and plant specific guidance.	Y[(2), (3), (4)]
<u>Suppression Pool Cooling Mode of RHR</u> The RHR system heat exchangers are placed on-line by the operator to maintain the containment within specific pressure and temperature boundary conditions prescribed in the EOPs. Containment heat removal affects both the magnitude and timing of a potential source term release to the environment. Timing (and magnitude) of an impending release can be extended by controlling containment pressure below the point at which structural failure occurs; whereas, the magnitude of the release can be affected by two phenomena: 1. maintaining the suppression pool temperature less than the NPSH and vortex limits of ECCSs taking suction off the pool; and 2. controlling suppression pool water temperature below saturation.	Level 2 IPE Tier 2 Document	4.6-10	N/A	Included in EPG/SAG and SAMP-1 and SAMP-2	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
<u>Suppression Pool Cooling Mode of RHR</u> (cont'd) Each of these phenomena are briefly discussed below. Maintaining suppression pool water temperature as low as possible extends the time that the operator can establish makeup to either the RPV or the drywell upon its breach. MAAP calculations have shown that the availability of water to cool debris (given that in-vessel recovery was unsuccessful) reduces both the impact to the containment, as well as the source term that accumulates inside the drywell air space. The suppression pool water temperature also affects the potential for "scrubbing" aerosols if the source term is directed through the pool before egress from the containment. MAAP calculations [Rev. 7.01] indicate that there is a correlation (i.e., inverse relationship) between the water temperature and the effective pool DF. Presently, these analyses indicate that the suppression pool is ineffective for scrubbing radionuclide aerosols once the water temperature achieves its saturation temperature. This assumption does not appear consistent with NEDO-24250 and recent experiments. In fact, due to bubble dynamics in a saturated pool, the DF may actually increase at saturation. It is the judgment of the IPE team that a DF of at least 10 for a saturated pool is reasonable. The MAAP results will be adjusted accordingly based on this judgment. Of course, this adjustment will only apply to the pool scrubbing portion of the source term for events with late drywell failure.	Level 2 IPE Tier 2 Document				

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
Water Injection Post Containment Failure From MAAP calculations [Rev. 7.01], it appears that the impact of continued water injection to the RPV or drywell post containment failure or venting (node MU of the CET) can be considered to have two possible effects: • For cases with drywell head failures it is found that the reduction in total CsI radionuclide release to the environment is reduced at most by approximately a factor of 2. Therefore, it will be assumed that for all cases with drywell head failure that the status of MU will not result in a reduction in source term to the next lower magnitude. Exception to this are cases in which drywell sprays are available. Then reductions of 1 magnitude are possible. • For cases in which the containment failure is in the wetwell the availability of MU or post containment water injection to the RPV or drywell will result in minimizing the releases.	Level 2 IPE Tier 2 Document	4.6-13	N/A	Adequate as is and as modified by EPG/SAG.	N
Alternate Water Supply for Drywell Spray/Vessel Injection PBAPS has the capability to inject HPSW through the RHR system to provide spray or injection using the Conowingo pond or Emergency Cooling Tower as water supplies. This capability, although powered from onsite AC emergency power, is enhanced through the use of an electrical cross-tieing procedure during LOOP events. Procedural guidance exists for the use of fire water as a vessel injection source through the RHR system.	IPE Submittal	6-2	Y	Significant enhancement in use of external injection in SAMP-1 and SAMP-2 (HPSW, T-205, T-231, T-245).	N

Table 3.2-1

IPE/PSA INSIGHTS (LEVEL 2)

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same As IPE (Y/N)		
Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability Power - All eleven SRVs are provided with two redundant 125 VDC power supplies to the solenoid of each valve. The containment isolation valves that provide long term nitrogen supply to the ADS valves are powered from emergency buses. The enhanced LOOP procedure also recognizes the need to provide emergency power to these valves. Nitrogen - The normal nitrogen supply to the ADS valves is backed by bottles and an outside connection for long term nitrogen supply. Procedures exist to manually bypass the containment isolation valves for the backup bottles to the ADS valves. Additionally, procedural guidance also exists to use the CAD tank as a supply of nitrogen to all the SRVs.	IPE Submittal	6-2	Y	No Additional Actions Necessary	N
Emergency Procedures and Training Peach Bottom has revised its EOPs to include the guidance of the BWROG EPG Revision 4. Substantial improvement in the LOOP procedure which addresses the prioritization of equipment and considers the inter-unit interactions was recently completed. Training is currently is progress. This procedure was credited in the analysis.	IPE Submittal	6-2	Y	N/A	N

Notes to Table 3.2-1:

(1) Containment Flooding for certain severe accidents (with RPV Breach)

A potential change that can be considered to the SAGs relates to the mitigation strategy proffered by TAB A - (Leg 1). This strategy involves the desire to flood containment to above TAF. This approach results in compressing the gas in the containment and causing the wetwell pressure to rise above the PCPL. This means that TAB A strategy currently forces containment venting. This can occur in 5 hours given typical MAAP RPV breach timings.

Recommended Change

Avoid containment venting until as late as possible in TAB A (Leg 1). This change would advocate covering debris with water but not continuing to flood to above TAF where venting would be required.

This would be a deviation from EPG/SAG and should be decided generically.

Potential SAG Change 2: RPV Vent

The AMWG restriction on RPV venting during DBA LOCA recovery has been removed from the SAGs. This is considered an important change. Options to address DBA issues include:

- Re-insert A restriction on RPV venting when in Leg 3 (e.g., DBA) -- RPV venting should be restricted to be used only if offsite releases are acceptable (< 10 CFR 100)
- Train personnel to switch to Leg 2 when recovering from DBA. This would limit the RPV venting and likely not require it.

This issue is addressed in the bases of PBAPS SAMP 1 and is covered in training.

(2) The drywell vent is currently not restricted in the EPG/SAG implementation. The prioritization in T-200 of the vent pathway type and size are not pre-defined. The options to use which vent pathways has been deferred to the operating staff. This may require training and possibly guidance in the TSGs to provide a method of minimizing releases while maximizing the other benefits of containment venting.

For this same situation, the new EPG/SAGs would not result in the operators venting the RPV. And as discussed before, the operators have greater flexibility to choose the appropriate venting strategy. Should the operators choose to vent the suppression chamber for this scenario, the releases would be expected to be in the range of 1%, similar to the scenario evaluated in the Peach Bottom IPE.

Venting implementation has been thoroughly discussed and considered in the EPG/SAGs. The results are to allow the maximum plant specific flexibility. This flexibility is now established at PBAPS in revised T-200 and TSG 3.3.

Footnotes to Table 3.2-1 (cont'd)

(3) Containment venting flexibility has been increased in the EPG/SAG guidance. This flexibility may require additional training or guidance to the operating staff or TSC personnel who will make containment venting decisions. The issues and topics related to containment venting that are of interest include the following:

Venting Timing

- Primary Considerations
  - Off-site Personnel Safety Maximized (includes Emergency Plan Implementation)
  - On-Site Personnel Safety Assured
  - Containment Structural Capability Adequate
  - Containment Vent Valve Capability Adequate
  - Preserve System Operability
    - SRV
    - EQ in drywell
    - RCIC
    - MSIV
    - LPCI
  - Containment Flooding Evolution
- Secondary Conditions
  - Containment Leakage Effects
  - Deinerting
  - Depletion of Non-condensibles
  - Loss of NPSH
  - Habitability and Accessibility

Vent Path Selection

- Power available
- Ease of local operation if required
- Time available for operation
- Size adequate
- Scrubbed or filtered release
- Path can be closed
- Path can be throttled
- Path can be monitored
- Path does not affect reactor building accessibility or habitability
- Path acceptable for on-site personnel

Footnotes to Table 3.2-1: (cont'd)

Vent Technique

- Minimizing the source term released
- Minimizing the use of the drywell vent
- Minimizing release while supporting containment flooding
- Preserving the non-condensibles (including an inert containment)
- Minimizing containment leakage
- Maximizing combustible gas control effectiveness

This guidance is now incorporated at PBAPS in revised T-200 and TSG 3.3. The vent path selection is specifically covered in TSG 3.3.1.

(4) Note that since the initiation and preparation of the Draft of this study, PECO has drafted and instituted first revisions of the PBAPS Severe Accident Management Procedures (SAMPs) and Technical Support Guidelines (TSGs) (and have revised the EPG based TRIP procedures). These issues are now appropriately considered and addressed at PBAPS.

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Seismic Capacity Based on the results of the seismic margin reviews, the PBAPS structures (including the primary containment and primary containment isolation system), components and systems are typically rugged. The majority of designs were found to have adequate margin. Some equipment modifications (typically enhanced anchorage) and further evaluations were identified to minimize the threat of seismic induced failures on safe shutdown. In addition, distributed systems were reviewed at building transitions; observed configurations possess adequate flexibility to accommodate displacements.	IPEEE Submittal	3-76; 3-89; and 7-3 to 7-10	(1) N/A (2) Y	The PBAPS seismic capacity is governed by design basis criteria in the UFSAR and other design basis documents. In addition, procedure SE-5, Earthquake, directs appropriate actions following a seismic event, such as: • inspection of damage to equipment, tanks, and structures • inspections for leaks • testing of emergency diesel generators and fire water pumps Nonetheless, the following were identified in the IPEEE to further minimize seismic risk.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
Seismic Capacity (cont'd)				• Enhanced bracing/anchorage of MCCs, switchgear, transformers, panels, and instrument racks. Equipment numbers: - 00B97/98/99 - 20(30)B10/11/12/13 (and adjacent exformers) - 00B94/95/96 - 20(30)15/16/17/18 - 20(30)X133/150 - 20X30/31/32/33 - 30X31/33 - 00(30)X103 - 0AX26/0BX26/0CX26 - 20D21/22/23 - 30D22/23/24 - 20(30)C32/33 - 20C722A/B - 30C722B - 20C818/819 - DPS-20224-1/3	

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Seismic Capacity</u> (cont'd)				• Enhanced bracing/anchorage of cable trays and conduit in Rx building 195' and RW building 165;. • Enhanced bracing/anchorage of various valves. Equipment numbers: - MO-33-0498 - MO2-13-4487 - MO3-13-5487 - AO2(3)-03-33 • Enhanced bracing/anchorage of HVAC equipment. Equipment numbers: - 0A(B)V035/36 - 00F043 - 0AV034 - PO2-0223-1/3 - El. 165' Mechanical Equipment Room HVAC ducting.	

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Seismic Capacity (cont'd)				• Enhanced bracing/anchorage of cranes and other misc. equipment: - low voltage switchgear breakers hoists - medium voltage switchgear breaker hoists - yard gantry crane - diesel generator overhead crane controllers - TIC-30223 temperature controller - U3 HCUs - lights overhead of battery racks • Enhanced housekeeping surrounding the following components: - 00C29A/B/C/D - 20C124/30C124 - 20C139 No additional modifications to EPGs/SAGs or other plant procedures/or equipment are judged necessary to address this insight.	

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Seismically Induced Internal Flooding</u> Walkdowns of distributed systems such as piping and HVAC systems were performed in conjunction with the walkdowns associated with other classes of components. As part of the distributed systems walkdown, the Seismic Review Team reviewed each area for the presence of piping and the potential for seismic induced flooding. In areas where SPCL equipment is located, the fire suppression systems are either a nitrogen filled manual pre-action system, CO_2 system, or an open nozzle water curtain system. With one minor exception, the team concluded that seismic induced flooding effects are not a significant threat. The one exception concerned a sprinkler head above Inverter 20D37. No drip shield was present on the inverter. The team judged that flooding was not an issue but the inverter may be impacted by spray.	IPEEE Submittal	3-69 to 3-70	(1) N/A (2) Y	A modification was identified for installation of a drip shield to protect inventer 20D37 from inadvertent spray. No additional modifications to EPGs/SAGs or other to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Seismic-Induced Masonry Wall Failure</u> No outiers exist with respect to seismic-induced failure of masonry walls.	IPEEE Submittal	3-75	(1) N/A (2) Y	This insight reflects a positive feature of the PBAPS plant and procedures. The PBAPS seismic capacity is governed by design basis criteria in the UFSAR and other design basis documents. In addition, procedure SE-5, Earthquake, directs appropriate actions following a seismic event such as: • inspection of damage to equipment, tanks, and structures • inspections for leaks • testing of emergency diesel generators and fire water pumps No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
Seismic-Induced Fires No situations were found where flammable gas or liquid storage vessels could create a significant fire hazard due to a seismic event.	IPEEE Submittal	4-71	(1) N/A (2) Y	Refer to disposition above.	N
Seismic Induced External Flooding No issues with respect to seismic-induced failure of dams, levees, and dikes. Failure of the upstream Holtwood dam coincident with the Probable Maximum Flood is considered in the PBAPS design.	IPEEE Submittal	5-59	(1) N/A (2) Y	This insight reflects a positive feature of the PBAPS plant and procedures. The PBAPS seismic capacity is governed by design basis criteria in the UFSAR and other design gas is documents. In addition, procedure SE-5, Earthquake, directs appropriate actions following a seismic event such as: • inspection of damage to equipment, tanks, and structures • inspections for leaks • testing of emergency diesel generators and fire water pumps	

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Seismic Induced External Flooding (cont'd)				Also, SE-4, Flood, directs appropriate preventive and mitigative actions in response to rising river levels. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
Mercury Switches Mercury switches are a special concern since they can cause fire protection equipment to spuriously operate during a seismic event. The fire protection systems were reviewed to determine if mercury switches are used. The following mercury switches were found on the fire system: • PS-0294, Diesel driven fire pump (00P063) discharge pressure switch • PS-0296, Electric motor drive fire pump (00P064) discharge pressure switch • CO_2 System Panels 0AC215, 0BC215, 20C215, 30C215 Cardox relays	IPEEE Submittal	4-71 to 4-74; 7-13	(1) N/A (2) Y	As the fire pumps and piping would not be damaged following spurious actuation, and they would be available for normal operation after a couple minutes into a seismic event, no modifications to the fire pumps discharge mercury switches have been instituted or judged warranted. Given the various spurious actuations caused by seismic actuation of the Cardox mercury relays, the establishment of procedural controls has been recommended to mitigate the results of possible actuations.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Mercury Switches</u> (cont'd) • Water Suppression System Manual Pull Stations - U2 Rx Bldg Water Curtain System remote manual pull station - U3 Rx Bldg Water Curtain System remote manual pull station - U2 Turbine 2-9 Manual Pre-Action System manual pull stations • CO2 System Hose Stations - HR-AA3 U2 Turbine Building CO2 Hose Station Located on elevation 165'-0" - HR-AA4 U3 Turbine Building CO2 Hose Station Located on elevation 165'-0" With respect to the fire pump switches, a seismic event could cause either or both pumps to start spuriously which would result in the pumps running at dead. However, since there are relief valves on the discharge piping, no damage to the pumps would result. In addition, a seismic event could also temporarily prevent the pumps from starting. The seismic event that is postulated for PBAPS is expected to last less than a minute; after which, it is expected that the pumps would resume normal operation.				Given the potential for spurious actuation of the water curtains, the U2 and U3 Reactor Building Water Curtain system manual pull stations have been identified for modifications to replace them with either push-button stations or pull stations which do not rely on mercury switches. As seismic actuation of the CO2 hose stations would not result in discharge or release of CO2, no modifications have been instituted or judged warranted. No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Mercury Switches (cont'd) With respect to the Cardox relays, a seismic event could cause the relays to actuate the equipment they control, such as: Derby electrical damper release mechanisms, local area alarm horns, and fire alarm system annunciator windows and automatic code transmitters. Spurious actuation of these pressure switches would not result in actuation of the CO_2 suppression systems themselves, however, all auxiliary actions listed above would occur. With respect to the water suppression system pull stations, a seismic event could cause the mercury bulb switches in the pull stations to actuate. Spurious actuation of these pressure switches would result in actual application of water in the case of the U2&U3 water curtains. With respect to the CO_2 hose stations, they are designed such that when either of the two hose nozzles are removed from their holders by a fire brigade member, a CO_2 valve opens which charges both hose stations with CO_2 and provide a local alarm. A seismic event could cause the hose stations listed above to "charge" with CO2 by shaking them from their holders. However, charging of these hose stations would not result in discharge or release of CO_2.					

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
<u>Seismic degradation of Fire Suppression Systems</u> A walkdown revealed some conditions which could potentially result in degradation of fire suppression systems during a seismic event: • CO2 tanks 00S101, 20S101, 30S101, and 20S112 are not anchored to the floor. Should these storage tanks fail during a seismic event, the CO2 systems associated with them would be rendered inoperable. • CO_2 battery racks 20D411 and 30D411 located adjacent to tanks 20S101, and 30S101 have some potential seismic vulnerabilities (lack of spacers between batteries, lack of end rails, etc.) Should the battery racks fail during a seismic event the automatic features of the CO_2 systems associated with them would be rendered inoperable. However, manual capability would still be available provided the storage tanks and associated piping survive the seismic event.	IPEEE Submittal	4-74; 7-15	(1) N/A (2) Y	Modifications to provide additional restraints for CO_2 tank 00S101 have been recommended. In addition, an engineering evaluation has been requested to be informed to determine the impact of a seismically induced CO_2 release in the turbine building. If appropriate, pertinent plant modifications will be planned to mitigate the effects of seismic failure of CO_2 tanks 20S101, 30S101, and 20S112. No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
High Pressure Fire Induced Accident Sequences Similar to the results in the IPE, the unavailability of high pressure injection, either from fire-induced or random failures and combined with the failure to depressurize the reactor provide a lower limit on the fire risk calculations and affect the potential to screen some areas.	IPEEE Submittal	4-64	(1) Y (2) Y	The EPGs/SAGs recognize the importance of depressurization and it continues to be emphasized as a critical operator action in procedure and training. No modifications to EPGs/SAGs or other plant procedures (or equipment are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
Important Fire Areas The important (i.e., unscreened at end of analysis) fire areas in the PBAPS fire IPEEE analysis are: • 6N, U2 Reactor Building, North • 13N, U3 Reactor Building, North • 25, Control/Cable Spreading Room • 32, 4KV Switchgear Room • 34, 4KV Switchgear Room • 50R-2/4, Turbine Building U3 Wing Area • 50R-9a, Turbine Building 13.2 KV Switchgear Area	IPEEE Submittal	4-68	(1) N/A (2) Y	Identification of the Control Room, switchgear rooms, and the main floors of the Reactor Building as key fire areas is consistent with industry fire IPEEE analyses. However, a number of actions have been planned by PBAPS to minimize the fire risk associated with these (and other) fire areas, including: • enhance control of transient combustibles • enhance fire brigade awareness • upgrade fire compartment barriers • enhance procedures to allow specific operator actions No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
<u>Fire Induced LOOP or Loss of Containment Heat Removal</u> Reactor coolant inventory control, pressure control, and reactor heat removal were typically achieved because of the diverse systems or trains of systems that were available to fulfill the function. However, fires that impact offsite power and containment heat removal appear to have the largest effect when assessing the risk significance of individual fire area.	IPEEE Submittal	4-64	(1) Y (2) Y	The T-300 procedures provide direction regarding fire initiated plant transients. In addition, the enhanced SE-11 procedure provides direction regarding cross tie of buses. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>Control Room Fires</u> Credit in the IPEEE for quick and effective manual suppression of panel fires by the Control Room operators is key to minimizing Control Room fire-induced core damage risk. The probability of a fire occurring in the Control Room affecting a particular cabinet combined with the probability of failure to detect and suppress a fire is small. The combined probability of events needed to prevent the safe shutdown of the plant results in a probability that is acceptable. Although the Control Room is considered acceptable from a risk perspective, it remains as one of the non-screened compartments per the FIVE methodology.	IPEEE Submittal	4-68	(1) N/A (2) Y	The T-300 procedures provide direction regarding fire initiated plant transients. In addition, the enhanced SE-11 procedure provides direction regarding cross tie of buses. No modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
<u>External Floods</u> During the course of the analysis, it was noted that Technical Specification 3/4.12.A, which covers high river level, contains action statements to shutdown the plant using normal operating procedures when the river level reaches an elevation of 113 feet, and to manually scram the reactor and place the plant in the cold shutdown condition if the river level exceeds 114 feet. However, the operating floor of the circulating and service water pumps is at elevation 112 feet. The floor has open grating exposing this area to the water level in the circulating water bays, which is practically river level. This mean that, during an external flooding scenario, up to two feet of water could accumulate in the non-safety-related portion of the pump house prior to any necessary action.	IPEEE Submittal	5-6 to 5-62	(1) N/A (2) Y	A Technical Specification change request was initiated to revise the required actions to be taken at river levels of 111 feet and 112 feet, respectively. This provides sufficient time to respond to an external flood event and transfer to the emergency cooling water system. However, as a result of conversion to the Improved Technical Specifications, the requirements for high river level were relocated to Section 3.15 of the Technical Requirements Manual and both it and plant procedures have been changed to require actions at 111 feet and 112 feet. No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source			Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
	Document	Page	Same as (1) IPE (Y/N) (2) IPEEE (Y/N)		
<u>High Winds</u> The Category I structures at PBAPS are designed to wind forces from a tornado having a total velocity of 300 mph. While the PBAPS design criteria does not meet the strict letter of 1975 SRP criteria (e.g., 300 mph tangential velocity and 60 mph transitional velocity), the PBAPS IPEEE analysis shows that (when considering both wind effects, tornado missiles, and hazard frequency), the PBAPS design satisfies the intent of the SRP and NUREG-1407.	IPEEE Submittal	7-2	(1) N/A (2) Y	This insight reflects positive features of the PBAPS design. The PBAPS structural resistance to high wind challenges is governed by design basis criteria in the URSAR and other design basis documents (e.g., PBAPS design basis document P-T-07). No modifications to the EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N
<u>Transportation and Nearby Facility Accidents</u> The evaluation of nearby industrial, transportation, and military facilities focused on two principal elements of the Standard Review Plan criteria: • Identification of potential hazards in the site vicinity; and • Evaluation of potential accidents.	IPEEE Submittal	5-90	(1) N/A (2) Y	Creations of Special Event procedures to address these hazards may be pursued but are currently not judged necessary given the calculated low risk impact. As such, no modifications to the EPGs/SAGs or other plant procedures (or equipment) are judged necessary to address this insight.	N

Table 3.3-1

IPEEE INSIGHTS

Insight	Source: Document	Source: Page	Source: Same as (1) IPE (Y/N) (2) IPEEE (Y/N)	Insight Disposition Including EPG/SAG Applicability	Additional Action May be Req'd (Y/N)
Transportation and Nearby Facility Accidents (cont'd) The results of the evaluation demonstrated, in general, that the PBAPS design satisfies the SRP intent. However, potential accidents, including toxic chemical release from on-site storage or nearby transportation routes (e.g. Conrail rail line, potential aircraft crashes), do not comply with the SRP. Analyses performed to evaluate the hazard associated with these accidents demonstrated that highway, railway, and aircraft accidents do not present significant risk to PBAPS.					

Table 3.4-1

SUMMARY OF BWR PLANT "VULNERABILITIES" IDENTIFIED BY LICENSEES

Vulnerability	PBAPS Approach to Resolve Vulnerability	Additional Action May Be Required (Y/N)
BWR 1, 2 & 3 (Isolation Condensers)		
Failure of isolation condenser makeup from city water supply and diesel fire-water pump, resulting in isolation condenser failure.	Not Applicable.	N
Operator failure to initiate isolation condenser to prevent safety relief valves from lifting in station blackout	Not Applicable.	N
Operator failure to restore or maintain RPV level following various accident scenarios	Determined by PECO to not be a vulnerability at PBAPS. Implemented EPG/SAG.	N
Drywell steel liner melt-through by molten debris following core melt and RPV failure.	Determined by PECO to not be a vulnerability at PBAPS. Implemented EPG/SAG.	N
BWR 3 and 4		
Loss of 3 of 4 residual heat removal (RHR) loops (directly or through loss of RHR service water (RHRSW)) as a result of catastrophic failure of either 4.16 kV Alternating Current (AC) safety bus.	Not Applicable.	N
Delayed loss of power and heat sinks caused by loss of switchgear or Class 1E Panel Room HVAC.	Determined by PECO to not be a vulnerability at PBAPS.	N
Upon high suppression pool temperature procedures requires manual operator actions to bypass HPCI suction transfer to suppression pool. Also must bypass high exhaust pressure trips for HPCI and RCIC upon high containment pressure.	Determined by PECO to not be a vulnerability at PBAPS. Not Required. HPCI/RCIC NPSH curves have been developed to further support continued operation under degraded plant conditions, e.g., CST unavailability.	N

Table 3.4-1

SUMMARY OF BWR PLANT "VULNERABILITIES" IDENTIFIED BY LICENSEES

Vulnerability	PBAPS Approach to Resolve Vulnerability	Additional Action May Be Required (Y/N)
Failure of HPCI and condensate during an ATWS is followed by reactor depressurization. Automatic LPCI initiation and injection of full flow for 5 minutes follows. Without immediate flow control by the operator, severe power excursion will occur.	Determined by PECO to not be a vulnerability at PBAPS. Addressed in Training. PECO does not have capability via overrides to control flow within 5 minute timer, but the pumps can be turned off.	N
During loss of offsite power or station blackout, condensate storage tank (CST) keepfill function is lost; occurrence of water hammer could cause failure of suppression pool cooling, causing containment failure unless CST is available for injection. Failure of the fire main as an injection source during station blackout will also result in vessel and containment failure.	Determined by PECO to not be a vulnerability at PBAPS.	N
BWR 5 and 6		
None.	---	N

Table 3.4-2

SUMMARY OF COMMON PLANT IMPROVEMENTS IDENTIFIED BY LICENSEES

Area of Improvement	Specific Improvement	PBAPS Has Evaluated Insight and Taken Actions	Additional Action May Be Required (Y/N)
AC Power	• Add or replace diesel generators	NAN	N
	• Add or replace gas turbine generator	NAN	
	• Implement redundant offsite power capabilities	NAN	
	• Improve bus/unit cross-tie capabilities	I	
DC Power	• Install new batteries, chargers or inverters	NAN	N
	• Implement alternative battery charging capabilities	NAN	
	• Increase bus load shedding	I	
Coolant Injection Systems	• Replace emergency core cooling system (ECCS) pump motors with air-cooled motors.	NAN	N
	• Align LPCI or core spray to CST upon loss of suppression pool cooling	NAN	
	• Align firewater system for reactor vessel injection	I	
	• Revise HPCI and RCIC actuation or trip setpoints	NAN	
	• Revise procedures to inhibit the automatic depressurization system (ADS) for non-ATWS scenarios	I	
Decay Heat Removal (DHR) Systems	• Add hard-pipe vent	NAN	N
Support Systems	• Implement procedures and install portable fans for alternative room cooling upon loss of HVAC	NAN	N
	• Install temperature alarms in rooms to detect loss of HVAC	NAN	
	• Revise procedures and training for loss of support systems	I	

Table 3.4-2

SUMMARY OF COMMON PLANT IMPROVEMENTS IDENTIFIED BY LICENSEES

Area of Improvement	Specific Improvement	PBAPS Has Evaluated Insight and Taken Actions	Additional Action May Be Required (Y/N)
ATWS	• Revise training on mechanically bound control rods	I	N
	• Install automatic ADS inhibit for ATWS scenarios	NAN	
	• Install alternative boron injection system	NAN	
Internal Flooding	• Increase protection of components from flood effects	I	N
	• Revise procedure for inspecting the floor drain and flood barriers	NAN	
	• Conduct periodic inspections of cooling water piping and components	NAN	
	• Install water-tight doors	NAN	
ISLOCAs	• Review surveillance procedures involving isolation valves	I	N
	• Modify procedure to depressurize the RCS to reduce leakage	NAN	
	• Revise training to deal with ISLOCAs	NAN	

NAN = No Action Necessary
I = Included
D = Deferred

Section 4
CONCLUSIONS

NEI 91-04 identifies the steps needed to provide closure to severe accident management issues. The steps include an evaluation of plant specific insights from the IPE or PSA. This evaluation has been performed for PBAPS and the results are identified in Section 3.

Insights identified as not yet fully incorporated into the Accident Management Guidelines (i.e., EPG/SAGs) are summarized in Table 4-1.

Note that since the initiation and preparation of the Draft of this study, PECO has drafted and instituted first revisions of the PBAPS Severe Accident Management Procedures (SAMPs) and Technical Support Guidelines (TSGs) (and have revised the EPG based TRIP procedures). These issues are now appropriately considered and addressed at PBAPS.

Table 4-1

SUMMARY OF PSA INSIGHTS FOR WHICH ADDITIONAL ACTIONS COULD BE CONSIDERED

Insight	Additional Action May Be Req'd (Y/N)	Possible Action	Recommended Action
<u>Vent: Timing of Radionuclide Release</u> The timing of containment venting can influence the radionuclide release by: • Releasing material early in an accident scenario • Minimizing blowdown flow from the containment that would otherwise occur from an uncontrolled containment rupture These effects on radionuclide release magnitude are considered small compared to the effect assigned to timing of release.	Y(1)	Address vent pathway prioritization and timing in training, TSGs, or additional guidance documents.	Incorporate additional guidance in TSGs for use during severe accidents. The drywell vent is currently not restricted in the EPG/SAG implementation. The prioritization in T-200 of the vent pathway type and size are not pre-defined. The options to use which vent pathways has been deferred to the operating staff. This may require training and possibly guidance in the TSGs to provide a method of minimizing releases while maximizing the other benefits of containment venting. For this same situation, the new EPG/SAGs would not result in the operators venting the RPV. And as discussed before, the operators have greater flexibility to choose the appropriate venting strategy. Should the operators choose to vent the suppression chamber for this scenario, the releases would be expected to be in the range of 1%, similar to the scenario evaluated in the Peach Bottom IPE. Venting implementation has been thoroughly discussed and considered in the EPG/SAGs. The results are to allow the maximum plant specific flexibility. **< This flexibility is now established at PBAPS in revised T-200 and TSG 3.3. >**

Table 4-1

SUMMARY OF PSA INSIGHTS FOR WHICH ADDITIONAL ACTIONS COULD BE CONSIDERED

Insight	Additional Action May Be Req'd (Y/N)	Possible Action	Recommended Action
Vent: Suppression Pool Scrubbing of Element Suppression pool water temperature (i.e., degree of subcooling) may affect the characteristic of the pool to retain aerosols during the vent. It is postulated that as the bulk temperature of the pool approaches saturation temperature, the effective DF of the pool decreases. In fact, the surrogate MAAP calculations indicate that upon reaching saturation temperature, the pool DF becomes unity (i.e., all aerosol radionuclides pass through the pool).	Y(1)	Flexibility to vent at low pressures provided by SAGs. Decision to vent deferred to training and plant specific guidance.	Containment venting flexibility has been increased in the EPG/SAG guidance. This flexibility may require additional training or guidance to the operating staff or TSC personnel who will make containment venting decisions. The issues and topics related to containment venting that are of interest include the following: Venting Timing • Primary Considerations - Off-site Personnel Safety Maximized (includes Emergency Plan Implementation) - On-Site Personnel Safety Assured - Containment Structural Capability Adequate - Containment Vent Valve Capability Adequate - Preserve System Operability -- SRV -- EQ in drywell -- RCIC -- MSIV -- LPCI - Containment Flooding Evolution

Table 4-1

SUMMARY OF PSA INSIGHTS FOR WHICH ADDITIONAL ACTIONS COULD BE CONSIDERED

Insight	Additional Action May Be Req'd (Y/N)	Possible Action	Recommended Action
Vent: Suppression Pool Scrubbing of Element (cont'd)			Venting Timing (cont'd) • Secondary Conditions - Containment Leakage Effects - Deinerting - Depletion of Non-condensibles - Loss of NPSH - Habitability and Accessibility Vent Path Selection • Power available • Ease of local operation if required • Time available for operation • Size adequate • Scrubbed or filtered release • Path can be closed • Path can be throttled • Path can be monitored • Path does not affect reactor building accessibility or habitability • Path acceptable for on-site personnel

Table 4-1

SUMMARY OF PSA INSIGHTS FOR WHICH ADDITIONAL ACTIONS COULD BE CONSIDERED

Insight	Additional Action May Be Req'd (Y/N)	Possible Action	Recommended Action
<u>Vent: Suppression Pool Scrubbing of Element</u> (cont'd)			<u>Vent Technique</u> • Minimizing the source term released • Minimizing the use of the drywell vent • Minimizing release while supporting containment flooding • Preserving the non-condensibles (including an inert containment) • Minimizing containment leakage • Maximizing combustible gas control effectiveness **< This guidance is now incorporated at PBAPS in revised T-200 and TSG 3.3. The vent path selection is specifically covered in TSG 3.3.1. >**

(1) Note that since the initiation and preparation of the Draft of this study, PECO has drafted and instituted first revisions of the PBAPS Severe Accident Management Procedures (SAMPs) and Technical Support Guidelines (TSGs) (and have revised the EPG based TRIP procedures). These issues are now appropriately considered and addressed at PBAPS.

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