APPENDIX G
SEVERE ACCIDENT MITIGATION ALTERNATIVES (SAMA)

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G.1 METHODOLOGY

The methodology selected for this analysis involves identifying those SAMA candidates that have the highest potential for reducing core damage frequency and person-rem risk and determining whether or not the implementation of those candidates is beneficial on a cost-risk reduction basis. This process consists of the following steps:

- Identify potential SAMA candidates based on NRC and industry documents,
- Screen out Phase 1 SAMA candidates that are not applicable to the Peach Bottom Atomic Power Station (PBAPS) design or are of low benefit in Boiling Water Reactors,
- Extend the current Peach Bottom Probabilistic Safety Analysis (PSA) (PB99 Rev 1) results (an update to Ref. G.8-23) to include both radionuclide releases and the related consequences (a Level 3 analysis). This requires conversion of the PBAPS Level 2 PSA results into the format used in NUREG/CR-4551[1] and scaling the Level 3 output based on those Level 2 PSA results and the demographic information of the surrounding communities at the end of the license extension,
- Determine the maximum averted cost-risk that is possible based on the PBAPS PSA Level 3 results,
- Screen out Phase 2 SAMA candidates whose estimated cost exceeds the maximum possible averted cost-risk,
- Perform a more detailed analysis to determine if the remaining SAMA candidates are desirable modifications or changes. This is based on a comparison of the averted cost-risk associated with implementing the SAMA at the site and the cost required to perform the modification. If the averted cost-risk is greater than the cost of implementation, then the SAMA candidate is considered to be a beneficial modification.

The steps outlined above are described in more detail in the subsections of this appendix.

[1] This is a technical report summarizing the input into NUREG-1150. Both NUREG/CR-4551 and NUREG-1150 are analyses sponsored by the NRC.

G.2 LEVEL 3 PRA ANALYSIS

The SAMA evaluation relies on Level 3 PSA results to measure the effects of potential plant modifications. A Level 3 model was created for PBAPS as part of NUREG-1150 and NUREG/CR-4551 (Ref. G.8-1 and G.8-2, respectively); however, while the Level 1 and 2 PSA models have been updated and enhanced to continually reflect plant changes since the publication of these NUREGs, the Level 3 model has not been updated.

Version 1.5 of the MACCS code (Ref. G.8-3) was used to perform the PBAPS Level 3 PSA in NUREG/CR-4551. The analysis was performed specifically for Peach Bottom Unit 2 and includes data unique to that site. While that report provides thorough documentation of the Level 3 analysis, the results are not directly used in the PBAPS SAMA evaluation. Some of the characteristics of the site data have changed since the performance of NUREG/CR-4551 in 1990 and it is considered necessary to account for these changes prior to applying the evaluation to this analysis.

Severe accidents due to external events, such as fire and seismic events, were evaluated in response to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities". The fire analysis utilized the Fire Induced Vulnerability Evaluation (FIVE) methodology. The seismic analysis employed the seismic margins methodology. Insights from the PBAPS IPEEE studies have been incorporated and are considered in the SAMA tables.

There are no seismic or fire PSA models that can be used to perform either the baseline SAMA calculation or identify the change in risk that could be attributed to any proposed SAMA. It is judged appropriate to use the internal events PSA as a gauge to effectively describe the risk change that can be attributed to SAMAs.

G.2.1 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 population at the end of the license extension in 2034. For the purposes of this analysis, the 2034 population is estimated using a simple, linear growth approximation for the population density in the surrounding area.

Population data from Table 4.2-2 of NUREG/CR-4551 was extrapolated to 50 miles from the plant (assuming a linear growth in population density away form the plant). The 1990 population estimate was derived from US census data and used in conjunction with the 1980 estimate to determine the increase in population per year. Using the 1990 50-mile population as a starting point, the growth rate (assumed to be constant) was applied over 44 years to approximate the population at the end of plant life in 2034. The population data used for this estimate is shown in the Tables G.2-1 and G.2-2. Table G.2-1 provides the information presented in Table 4.2-2 of NUREG/CR-4551 and Table G.2-2 summarizes the 1990 US census information.

TABLE G.2-1
NUREG/CR-4551 POPULATION DATA

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

Table G.2-2 was developed using data available on the US Census Bureau's web site (http://www.census.gov). Population from the 1990 census is available by county and was used to estimate the population within the 50 mile radius of the plant. An atlas containing a mileage scale and county borders was used to identify the counties within the 50 mile radius. If the entire county fell within the 50 mile radius, then the entire population was included in the 50 mile estimate. Otherwise, a fraction of the population was counted based on the percentage of the county within the 50 mile radius. The land area within the 50 mile radius is estimated based on visual inspection of the map and the population of that area is estimated assuming uniform distribution of the population within the county. The results are presented below:

TABLE G.2-2
POPULATION WITHIN 50 MILES OF PBAPS (1990 US CENSUS)

County Name	Total Population	Percent Included Within 50 Miles of PBAPS	Population within 50 Miles of PBAPS
Delaware	547651	85%	465503.35
Montgomery	678111	15%	101716.65
Berks	336523	50%	168261.5
Lebanon	113744	75%	85308
Adams	78274	40%	31309.6
Dauphin	237813	40%	95125.2
Cumberland	195257	10%	19525.7
Carroll	123372	85%	104866.2
Queen Anne's	33953	60%	20371.8
Anne Arundel	427239	30%	128171.7
Howard	187328	50%	93664
Salem	65294	50%	32647
Gloucester	230082	20%	46016.4
Kent, DE	110993	25%	27748.25
York	339574	100%	339574
Lancaster	422822	100%	422822
Chester	376396	100%	376396
Baltimore	692134	100%	692134
Baltimore City	736014	100%	736014
Harford	182132	100%	182132
Cecil	71347	100%	71347
Kent, MD	17842	100%	17842
New Castle	441946	100%	441946
		Total =	4700442.35

The actual number used in the SAMA calculations to adjust the NUREG/CR-4551 results is a ratio of the population density for the area within 50 miles of the plant in the year 2034 to that in 1980. This ratio, $P_{34/80}$, is calculated as follows:

$$P_{34/80} = \left(\frac{\frac{\left(PD_{50(1990)} - PD_{50(1980)}\right)}{(1990-1980)} * 44\,years + PD_{50(1990)}}{PD_{50(1980)}} \right)$$

Where:

$P_{34/80}$ = Ratio of the population density for the area within 50 miles of the plant in 2034 to the population density for the area within 50 miles of the plant in 1980

$PD_{50(1990)}$ = Population density for the area within 50 miles of the plant in 1990 (based on 1990 US census data)

$PD_{50(1980)}$ = Population density for the area within 50 miles of the plant in 1980 (based on NUREG/CR-4551)

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

$P_{34/80}$ is used to scale the Population Dose Risk (PDR) within 50 miles to reflect the population characteristics of the site area at the end of the proposed life extension. This affects the Offsite Exposure Cost Risk and the Offsite Economic Cost Risk used in the determination of the Baseline Screening Cost and the averted cost-risk for any proposed SAMAs.

Applying census data for the area around PBAPS results in the following:

$$P_{34/80} = \frac{\left[\frac{(598.5-385)}{(1990-1980)} * 44 + 598.5\right]}{385} = 3.99$$

G.2.2 ECONOMY AND AGRICULTURE

As part of NUREG/CR-4551, site specific data were collected on the economic and agricultural characteristics surrounding the Peach Bottom site. It is assumed that the relative distribution of these factors has remained constant and that the overall growth in "economy" and "agriculture" is represented by the growth in population. This growth is reflected by means of scaling the Offsite Economic Cost Risk by the increase in population.

G.2.3 OTHER PLANT SPECIFIC DATA

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

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

G.2.4 CONVERSION OF PBAPS PSA MODEL RESULTS TO LEVEL 3 OUTPUT

A major factor related to the use of NUREG/CR-4551 in the SAMA evaluation is that the PBAPS PSA has been enhanced to reflect plant changes and new information. While consistent with, the Individuals Plant Examination (IPE), the level of sophistication of the PSA model has increased and the results have changed as modeling techniques have improved. In addition, the results of the PBAPS PSA Level 2 model are not defined in the same terms as reported in NUREG/CR-4551. In order to use the Level 3 model presented in that document, it was necessary to convert the PBAPS PSA Level 2 model results into a format which allowed for the scaling of the Level 3 results based on current Level 2 output. Finally, as mentioned above, the Level 3 results were modified to reflect the expected change in the site demographics at the end of the proposed license extension. This subsection provides a description of the process used to convert the PBAPS PSA Level 2 model results into a form that can be used to generate Level 3 results using the NUREG/CR-4551 documentation. The Unit 2 PSA model, which has a slightly higher core damage frequency (CDF) between the Unit 2 and Unit 3 models, is used for the calculations in this study. Figure G.2-1 provides a graphical reference of the steps taken in NUREG/CR-4551 to determine the offsite consequences (Level 3 results) based on Level 1 analysis input (Plant Damage State frequencies).

G.2.4.1 Identification of Required Parameters

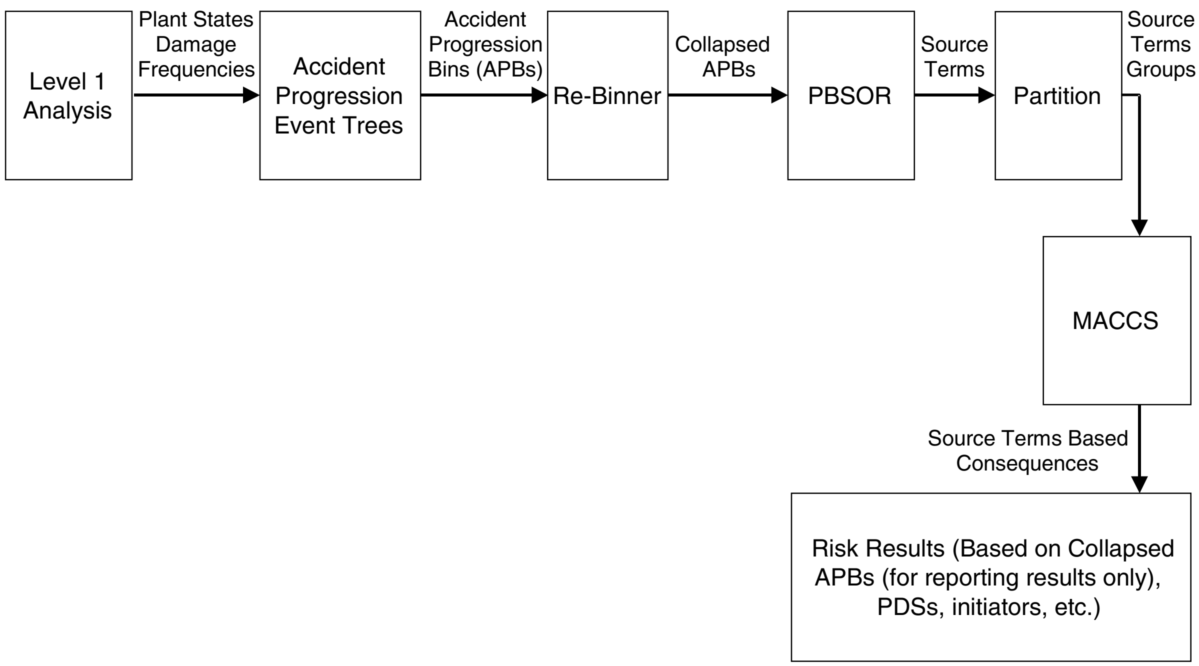
The first step in the conversion of the PBAPS PSA results into a format suitable for updating the NUREG/CR-4551 Level 3 results is to identify the output of the Level 3 model that is required in the cost-benefit calculations, which are described in Section G.3. While the CDF from the Level 1 model is used in these calculations, there are specific Level 3 terms that are needed to complete the analysis. Determination of the Offsite Exposure Cost Risk and the Offsite Economic Cost Risk both require Level 3 input. Offsite Exposure Cost Risk requires an estimate of the Population Dose Risk (0-50 miles) and the Offsite Economic Cost Risk requires the economic cost of an accident. Subsections G.2.4.2 and G.2.4.3 describe how these results are obtained, respectively.

G.2.4.2 Determination of Population Dose Risk (0-50 Miles)

The basic process that was pursued to obtain Level 3 results based on the PBAPS PSA Level 2 model and NUREG/CR-4551 was to define a useful relationship between the Level 2 and Level 3 results. NUREG/CR-4551 defines the fractional contribution of the 10 collapsed Accident Progression Bins (APBs) to the Population Dose Risk at 50 miles (PDR50). It was also determined that the frequency of each collapsed APB could be calculated based on the information provided in NUREG/CR-4551. Given this relationship, it was possible to determine the PDR50 based on the results of the PBAPS PSA model if those results are reported in terms of the same accident bins. For example, for a given collapsed APB:

$$\mathrm{PDR50}_{(\mathrm{PBAPS\,PSA})} = \frac{\mathrm{PBAPS\,PSA\ Frequency}}{\mathrm{NUREG/CR-4551\ Frequency}} * \mathrm{Collapsed\ APB\ Fractional\ Contribution} * \mathrm{Total\ PDR50}_{(\mathrm{NUREG/CR-4551})}$$

Figure G.2-1
NUREG/CR-4451 Level 2 and 3 Process Summary



If this is performed for each of the 10 collapsed APBs and the results are summed, the total is the PDR50 for the PBAPS PSA. In the determination of Offsite Exposure Cost Risk, however, the PDR50 should reflect the site conditions at the end of the renewed license term in 2034 (conservative). This is calculated by scaling the PDR50 results for the PBAPS PSA model by the $P_{34/80}$ ratio to account for the change in population. Table G.2-3 summarizes the results of this process.

TABLE G.2-3
CALCULATION OF PDR50

Collapsed Bin #	Fractional APB Contributions to Risk (MFCR)[1]	NUREG/CR-4551 Population Dose Risk at 50 miles (From a total of 7.9 person-rem, mean)[2](MFCR)	NUREG/CR-4551 Collapsed Bin Frequencies[3] (per year)	PBAPS PSA Collapsed Bin Frequencies[4] (per year)	PBAPS PSA Population Dose Risk at 50 miles (MCFR) (1980 Pop Data)[5] (person-REM)	Population Dose Risk at 50 miles (MCFR) (PBAPS PSA, scaled to 2034 population using $P_{34/80}$) (person-REM)
1	0.021	0.1659	9.55×10^{-8}	0	0.00	0.00
2	0.0066	0.05214	4.77×10^{-8}	0	0.00	0.00
3	0.556	4.3924	1.48×10^{-6}	4.66×10^{-8}	1.38×10^{-1}	5.52×10^{-1}
4	0.226	1.7854	7.94×10^{-7}	1.42×10^{-6}	3.19	1.28×10^{-1}
5	0.0022	0.01738	1.30×10^{-8}	1.17×10^{-7}	1.56×10^{-1}	6.24×10^{-1}
6	0.059	0.4661	2.04×10^{-7}	2.01×10^{-9}	4.59×10^{-3}	1.83×10^{-2}
7	0.118	0.9322	4.77×10^{-7}	2.25×10^{-8}	4.39×10^{-2}	1.75×10^{-1}
8	0.0005	0.00395	7.99×10^{-7}	1.42×10^{-8}	7.02×10^{-5}	2.81×10^{-4}
9	0.01	0.079	3.86×10^{-7}	7.38×10^{-7}	1.51×10^{-1}	6.03×10^{-1}
10	0	0	4.34×10^{-8}	0	0.00	0.00
				Totals	3.69	14.72

Notes to Table G.2-3:

1. From Table 5.2-3 of NUREG/CR-4551

2. The total population dose risk at 50 miles from internal events in person-rem is provided in Table 5.1-1 of NUREG/CR-4551. The contribution for a given APB is the product of the total PDR50 and the fractional APB contribution.

3. NUREG/CR-4551 provides the conditional probabilities of the collapsed APBs in Figure 2.5-6. These conditional probabilities are multiplied by the total internal CDF to calculate the collapsed APB frequency.

4. Determined by re-grouping PBAPS PSA results into the 10 collapsed APBs.

5. This column is the ratio of the PBAPS PSA collapsed APB frequency to the NUREG/CR-4551 collapsed APB frequency multiplied by the NUREG/CR-4551 APB specific PDR50 contribution.

Each sequence of the PBAPS PSA Level 2 model was reviewed and re-categorized into one of the collapsed APBs. The Level 2 model contains a significantly larger amount of information about the accident sequences than what is used in the collapsed APBs in NUREG/CR-4551 and the re-categorization required simplification of accident progression information and assumptions related to categorizations of certain items.

The collapsed APBs are characterized by 5 attributes related to the accident progression. Unique combinations of the 5 attributes result in a set of 10 bins that are relevant to the analysis. Information from the PBAPS PSA Containment Event Trees (CETs) was used to classify each of the Level 2 sequences using these attributes. The definitions of the 10 collapsed APBs are provided in NUREG/CR-4551 and are reproduced in Table G.2-4 for references purposes:

TABLE G.2-4
COLLAPSED APB DESCRIPTIONS

Collapsed APB Number	Description
1	CD, VB, Early CF, WW Failure, V Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
2	CD, VB, Early CF, WW Failure, V Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the wetwell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).
3	CD, VB, Early CF, DW Failure, V Pressure > 200 psi at VB Core damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is greater than 200 psi at the time of vessel breach (this means DCH is possible).
4	CD, VB, Early CF, DW Failure, V Pressure < 200 psi at VB Core Damage occurs followed by vessel breach. The containment fails early in the drywell (i.e., either before core damage, during core damage, or at vessel breach) and the RPV pressure is less than 200 psi at the time of vessel breach (this means DCH is not possible).

TABLE G.2-4 (Cont'd)
COLLAPSED APB DESCRIPTIONS

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

CD = core damage	CF = containment failure	DCH = direct containment heating
DW = drywell	MCCI = mollen concrete interaction	RPV = reactor pressure vessel
VB = vessel research	vent = venting	WW = wetwell

Some general assumptions were made during the classification of the Level 2 CET sequences in order to categorize certain sequences that contained characteristics that did not directly fit into one of the 10 collapsed APBs. As it is possible for these assumptions to vary between each of the 5 accident classes, each accident class is associated with a unique set of assumptions on a node by node basis. The “nodes” in the CETs represent phenomenological events, operation of plant systems, and operator performance. Table G.2-5 summarizes the accident class definitions and Table G.2-6 summarizes the nodal assumptions used to group the PBAPS PSA Level 2 sequences into the collapsed bins.

TABLE G.2.5
ACCIDENT CLASS DEFINITIONS

Accident Class Designator	Definition
1A	Accident Sequences involving loss of inventory makeup in which the reactor pressure remains high
1B	Accident sequences involving a loss of offsite power and loss of inventory makeup.
1C	Accident sequences involving a loss of inventory makeup induced by an ATWS sequence.
1D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully collapsed to 200 psi. Accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.
1E	Accident sequences caused by common mode failures that result in multiple front line system failures with the reactor at high pressure.
2A	Accident sequences involving a loss of containment heat removal and no venting capability.
2F	Accident sequences involving a loss of containment heat removal and no venting capability.
2T	Accident sequences involving a loss of containment heat removal and no venting with injection terminated prior to containment failure.
3A	Accident sequences leading to core vulnerable conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.
3B	Accident sequences initiated by or resulting in small or intermediate LOCAs for which the reactor can not be depressurized.
3C	Accident sequences that are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate challenging containment integrity.
4A	Accident sequences involving a failure to insert negative reactivity leading to a containment vulnerable condition due to high containment pressure.
5	Unisolated LOCA outside containment.

TABLE G.2.6
NODAL ASSUMPTIONS

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
1	IS – Containment Isolation	If the containment is not isolated, it is assumed that it will be open for the equivalent of an un-scrubbed release as soon as the vessel is breached. No depressurization is asked prior to this node; it is assumed that RPV pressure is >= 200 psi for these sequences. This is bin #3.
1	OP – Operator depressurizes the RPV	It is assumed that success on this branch results in RPV pressure below 200 psi.
1	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9. However, there is one case in which combustible gas venting (GV) fails followed by containment failure (CZ); this is assumed to result in a high early release and is categorized as a bin #4 event for low pressure and #3 for high pressure.
1	CX – Containment Intact During Flood, RPV Breach	Failure of containment during flood is assumed to result in an un-scrubbed release. The timing is technically later than vessel breach, but it is conservatively assumed to be “early” and is grouped in bins 3 or 4 depending on RPV pressure.
1	NC – No Large Containment Failure	A large containment failure instigated by high containment pressure following vessel breach is assigned to the “late containment failure” bins. The sequences contributing to these bins need to be separated into either WW or DW failures. While the PB CETs distinguish between these types of failures, the NUREG/CR-4551 analysis appears to take credit for scrubbing for any WW release (with respect to the collapsed bins in section 2.4.3). Not all WW failure in the CETs can be credited with successful scrubbing. Given a large containment failure, the only successful scrubbing path is that in which the WW fails in an area above the water level (success in node WW).
1	MU – Coolant Inventory Makeup	Coolant inventory makeup is assumed only to provide cooling to the core debris. No credit is taken for any potential scrubbing effects that water coverage may yield.
1	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub. This is judged to be conservative.
2	RX – Core Melt Arrested in Vessel	A success on this branch signifies that there is no vessel breach. The sequences following this path are grouped in bin #9. However, For accident class 2T sequences in which core melt has been mitigated in the vessel, a failure in the CZ node is also assumed to result in bins 3 or 4 according to RPV pressure. Given that there is no vessel breach, this is judged to be conservative.

TABLE G.2.6 (Cont'd)
NODAL ASSUMPTIONS

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

TABLE G.2.6 (Cont'd)
NODAL ASSUMPTIONS

Accident Class	PBAPS PSA Containment Event Tree Node	Assumption
3	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
4	RB – Release Mitigated in Reactor Building	The RB node, release mitigated in reactor building, is not credited as a scrubbing mechanism. The only scrubbing accounted for in the collapsed bins is distinguished by indicating a WW release and the amount of scrubbing that the reactor building is capable of providing is not considered to be the equivalent a WW scrub. This is judged to be conservative.
4	SP – Suppression Pool Not Bypassed	The suppression pool bypass node is considered in the PB CETs to determine whether the vent volume passes through the suppression pool or not. This node is quantified in Class 4 accidents for only "no breach" cases. For no vessel breach Bin #9 is assigned.
4	CZ/SI – Containment Intact/Mark I Shell Failure	Given that the core melt has not been contained in the RPV, failure in node CZ is assumed to result in an un-scrubbed release through the drywell. Failure in node SI is also assumed to result in an un-scrubbed release due to fission product release through the gap between the liner and the concrete. No credit is given to reactor building scrubbing (RB) or to injection to the DW or RPV (TD). The sequences with failures in these nodes are assigned to bins 3 or 4 depending on RPV pressure.
5	N/A	No collapsed bin is available for containment bypass scenarios. The closest match to a bypass scenario is assumed to be a vessel breach with early drywell failure (bins 3 and 4). These bins are assigned based on RPV pressure (failure to depressurize is set to 0.0, so all sequences with non-zero results will be assigned to bin #4).

G.2.4.2.1 Summary

The complete results of the Level 2 re-categorization are not presented here as there are over 1900 sequences in the CETs. Refer to Table G.2-3 for the collapsed bin frequencies calculated for the PBAPS PSA model. The APBs with the most influence on the PDR50 are 3, 4, and 7. The frequency for APB 3 dropped by about 2 orders of magnitude and as a result, this bin is no longer the dominant contributor to the PDR50. Conversely, the frequency of bin 4 increased by a factor of 2 and the bin now contributes about 87% of the PDR50. APB 7 was collapsed in frequency by a factor of 5 and remains as a significant, but non-dominant contributor to the results. It is also important to note that there were no Level 2 sequences categorized in APBs 1 or 2. This is primarily due to the assumption that failure on the SI node (shell melt through) results in an un-scrubbed release. The collapsed APBs treat a wetwell release as a scrubbed release, thus, the SI failures (this node is 1.0) are binned with the drywell failures to prevent un-scrubbed sequences from being categorized with the scrubbed releases. An early failure of containment due to the effects of vessel breach (CZ) is also assumed to result in an un-scrubbed release and therefore is not binned in APBs 1 or 2. This is judged to be conservative.

The end result is a baseline PDR50 of 14.7 person-rem per year per plant based on the scaled population data for 2034.

G.2.4.3 Determination of Offsite Economic Cost Risk

The Offsite Economic Cost Risk (OECR) results for the PBAPS PSA model depend on the relationship between the collapsed APBs and the Plant Damage States (PDSs) defined in NUREG/CR-4551. Plant damage states are groups of sequences that behave similarly in the Level 2 analysis; their descriptions are reproduced from NURGE/CR-4551 for reference purposes in Table G.2-7.

TABLE G.2-7
PLANT DAMAGE STATE DEFINITIONS

Plant Damage State Number	Description
1	(LOCA) This PDS is composed of two accident sequences: the first is a large LOCA followed by immediate failure of all injection; the second is a medium LOCA with initial HPCI success but almost immediate failure as the vessel depressurizes below working pressure, all other injection has failed. Early core damage results. CRD and containment heat removal are working. Venting is available.
2	(Fast Transient, SORV, RHR avail.) This PDS is composed of four sequences consisting of a transient initiator followed by two stuck open SRVs (the equivalent of an intermediate LOCA). HPCI works initially, but fails when the vessel depressurizes below HPCI working pressure; all other injection has failed and early core damage results. CRD and containment heat removal are working as in PDS 1 but steam is directed through the SRVs to the suppression pool and not to the drywell as in PDS 1. Venting is available.

Table G.2-7 (Cont'd)
Plant Damage State Definitions

Plant Damage State Number	Description
3	(Fast Transient, SORV, RHR not avail.) This PDS is similar to PDS-2 except that the containment heat removal is not working and CRD may not be working for some sub-groups (however, CRD is assumed to be working since the cutsets where it is not are negligible contributors).
4	(Fast Blackout) This PDS is a short term station blackout with DC power failed. It consists of 2 sequences: one with a stuck open SRV and one without a stuck open SRV. Early core damage results from the immediate loss of all injection. Venting is possible if AC power is restored (manual venting is possible if AC is not restored but considered unlikely).
5	(Slow Blackout) This PDS is a long term station blackout. It is composed of three sequences, one of which has a stuck open SRV. High pressure injection is initially working. AC power is not recovered and either: 1) the batteries deplete, resulting in injection failure, reclose of the ADS valves, and re-pressurization of the RPV (in those cases where an SRV is not stuck open), followed by boiloff of the primary coolant and core damage, or 2) HPCI and RCIC fail on high suppression pool temperature or high containment pressure, respectively, followed by boiloff and core damage at low RPV pressure (Since DC has not failed, ADS would still be possible, or an SRV is stuck open). The containment is at high pressure but less than or equal to the saturation pressure corresponding to the temperature at which HPCI would fail (i.e., about 40 psig at the start of core damage).
6	(Fast ATWS, SLC avail.) This PDS is an ATWS with SLC working. HPCI works and the vessel is not manually depressurized. Injection fails on high suppression pool temperature and early core damage ensues. Venting is available.
7	(ATWS, SORV) This PDS is an ATWS with failure of SLC; the initiator is a stuck open SRV. Otherwise, it is the same as PDS 8.
8	(ATWS) This PDS is an ATWS sequence with loss of an AC bus or PCS followed by failure to scram. High pressure injection fails on high suppression pool temperature and the reactor is either: 1) not manually depressurized or 2) the operator depressurizes and uses low pressure injection systems until the injection valves fail due to excessive cycling or, containment fails or is vented and the injection systems fail due to harsh environments in the reactor building or loss of NPSH (condensate cannot supply enough water since the CST can only supply about 800 gpm to the condenser. Condensate can only last a few minutes.). Early core damage ensues in case 1 and late core damage in case 2. Venting will not take place before core damage if the operator does not depressurize; but, it may, if he goes to low pressure systems. RHR and CSS are working and the containment pressure will begin to drop in case 1 or will level off at the venting or SRV reclosure pressure in case 2.
9	(ATWS, LOSP) This PDS is an ATWS with failure of SLC, the initiator is T1 (LOSP); however, other AC is available. Otherwise, this PDS is the same as PDS 8.

As there is no direct relationship documented between the collapsed APBs and the OECR, it was necessary to develop this relationship. This relationship allowed for the calculation of PBAPS PSA PDS frequencies based on the PBAPS PSA collapsed APB frequencies (the collapsed APB frequencies developed for the PDR50 calculation were also implemented here). A ratio of the PBAPS PSA PDS frequencies to the NUREG/CR-4551 frequencies multiplied by the NUREG/CR-4551 PDS OECR contributions provided the OECR for the PBAPS PSA model. The result was modified to account for the increased population at the end of the license (2034) as it was for the PDR50. The following steps summarize the process used to calculate the OECR for the PBAPS PSA:

1. Using Table C-1 of NUREG/CR-4551, calculate the OECR for each source term by multiplying the mean source term frequency by the Economic Cost associated with the source term.

2. Sum the source term specific OECR values to get a total OECR for the NUREG/CR-4551 analysis.

3. Calculate the fractional contribution of each PDS to each collapsed APB from NUREG/CR-4551. This number is the fraction of the total collapsed APB frequency contributed by a given PDS.

4. Calculate the PDS frequencies for the PBAPS PSA. These are the sums of the products of the collapsed APB frequency and the fractional contribution of each PDS over all collapsed APBs for all PDSs.

5. Calculate the NUREG/CR-4551 PDS contributions to the OECR. This is the total NUREG/CR-4551 OECR multiplied by the fractional contribution of each PDS.

6. Multiply the PDS specific OECR by the ratio of the PBAPS PSA PDS frequencies to the NUREG/CR-4551 PDS frequencies to obtain the OECR for the PBAPS PSA.

Multiply the PBAPS PSA OECR by the P34/80 ratio to obtain the OECR for the Peach Bottom site in 2034. This represents the OECR for a single unit core damage accident (per year).

These steps are discussed in more detail below and are represented graphically in Figure G.2-2.

Steps 1 and 2

The information in Table C-1 of NUREG/CR-4551 is summarized in Table G.2-8. This table includes the source term group identifier, the mean frequency of the source term, the economic cost of a release of the source term to the environment, and the OECR for the source term, which is the product of the source term's mean frequency and its economic cost. The source term groups are the product of the PARTITION computer program. PARTITION receives the individual source terms from PBSOR and organizes them into groups in order to limit the number of calculations that MACCS is required to perform.

Figure G.2-2
PBABS PSA OECR Calculation Process

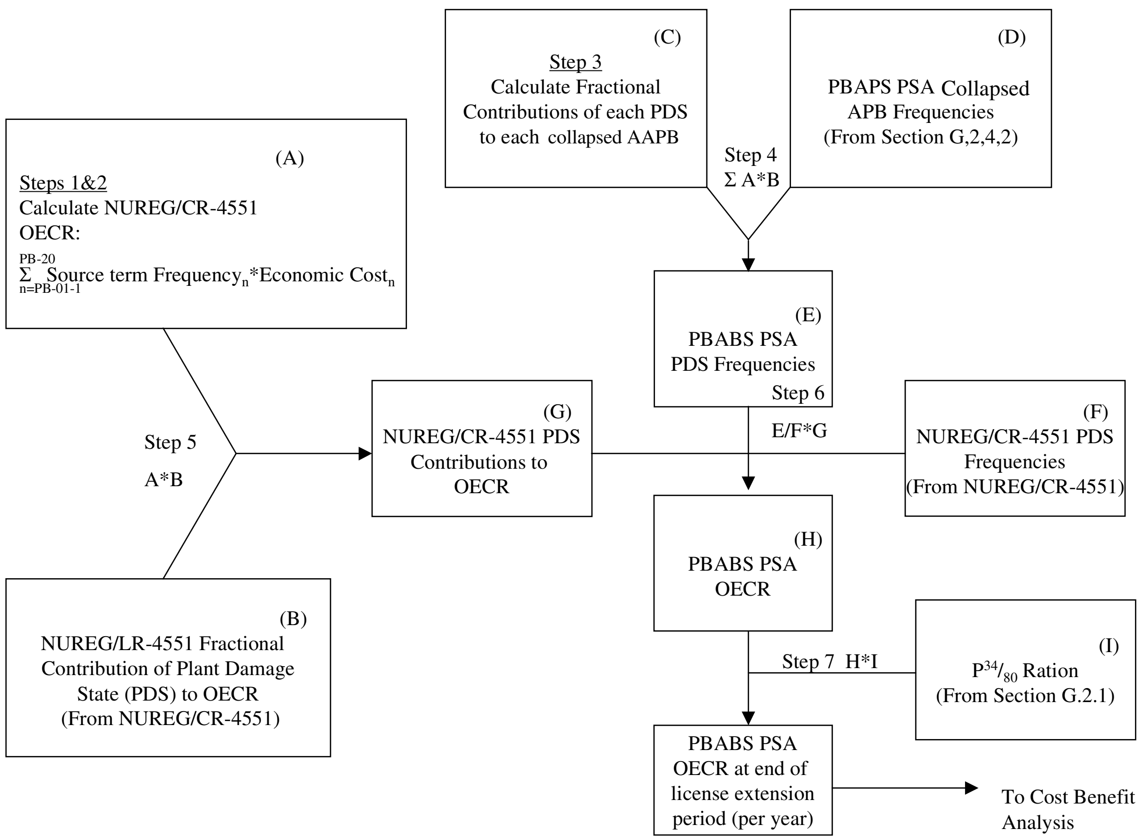


TABLE G.2-8
NUREG/CR-4551 OECR

Source Term Identifier	Mean Frequency	Economic Cost	NUREG/CR-4551 Annual Offsite Economic Cost-Risk (NUREG/CR-4551)
PB-01-1	1.00×10^{-7}	7.12×10^{7}	$7.12
PB-01-3	7.14×10^{-8}	6.99×10^{7}	$4.99
PB-02-1	5.26×10^{-8}	4.57×10^{8}	$24.04
PB-02-3	5.51×10^{-8}	5.01×10^{8}	$27.61
PB-03-1	1.15×10^{-7}	7.18×10^{8}	$82.57
PB-03-3	1.10×10^{-7}	3.11×10^{8}	$34.21
PB-04-1	9.73×10^{-8}	6.57×10^{8}	$63.93
PB-04-3	2.00×10^{-8}	6.32×10^{8}	$12.64
PB-05-1	8.38×10^{-8}	2.05×10^{9}	$171.79
PB-05-3	3.29×10^{-8}	1.36×10^{9}	$44.74
PB-06-1	1.28×10^{-7}	2.68×10^{9}	$343.04
PB-06-3	2.48×10^{-8}	2.43×10^{9}	$60.26
PB-07-1	3.25×10^{-7}	2.62×10^{9}	$851.50
PB-07-3	1.46×10^{-7}	2.36×10^{9}	$344.56
PB-08-1	7.52×10^{-8}	3.27×10^{9}	$245.90
PB-08-3	7.57×10^{-9}	1.10×10^{9}	$8.33
PB-09-1	7.56×10^{-8}	1.12×10^{10}	$846.72
PB-09-3	1.59×10^{-8}	7.31×10^{9}	$116.23
PB-10-1	1.67×10^{-7}	1.03×10^{10}	$1,720.10
PB-10-3	9.56×10^{-9}	7.30×10^{9}	$69.79
PB-11-1	1.90×10^{-7}	6.26×10^{9}	$1,189.40
PB-11-3	5.08×10^{-9}	4.54×10^{9}	$23.06
PB-12-1	5.66×10^{-8}	3.70×10^{10}	$2,094.20
PB-12-3	6.60×10^{-10}	3.60×10^{10}	$23.76
PB-13-1	2.49×10^{-7}	2.48×10^{10}	$6,175.20
PB-13-3	1.52×10^{-8}	2.50×10^{10}	$380.00
PB-14-1	6.08×10^{-7}	1.47×10^{10}	$8,937.60
PB-14-3	6.32×10^{-9}	1.62×10^{10}	$102.38
PB-15-1	1.59×10^{-9}	6.40×10^{10}	$101.76
PB-15-3	5.24×10^{-10}	6.37×10^{10}	$33.38
PB-16-1	4.28×10^{-8}	4.93×10^{10}	$2,110.04
PB-16-3	$1.19 \times 10\text{-}^{9}$	4.74×10^{10}	$56.41
PB-17-1	3.67×10^{-7}	3.67×10^{5}	$0.13
PB-18-1	6.94×10^{-7}	1.15×10^{6}	$0.80
PB-19-1	3.29×10^{-7}	3.49×10^{8}	$114.82
PB-19-3	2.48×10^{-8}	4.39×10^{7}	$1.09
		TOTAL=	$26,424.10

The total OECR for the NUREG/CR-4551 analysis is $26,424.10. The OECRs calculated for other plants, such as Edwin I. Hatch, are significantly lower than the estimate for PBAPS. This is primarily due to the demographics of the site areas.

Step 3

The next step in the process is to define the relationship between the PDSs and the collapsed APBs. Figure 2.5-5 of NUREG/CR-4551 provides the conditional probabilities for each PDS's contribution to each collapsed APB. These probabilities cannot be used to directly translate between the collapsed APB frequency and the PDS frequency because each PDS only provides a portion of the total collapsed APB frequency. It is necessary to calculate the fraction of the collapsed APB frequency contributed by each PDS. Once this is established, if a new collapsed APB frequency is provided, these fractions can be applied to each PDS and the new APB frequency can be distributed among all of the PDSs. If this is performed for each APB, the sum of the contributions from each APB to a given PDS can be summed to calculate the new PDS frequency. The first part of this process is defining the conditional probabilities for each PDS for each collapsed APB. As mentioned above, NUREG/CR-4551 Figure 2.5-5 provides these results. They are reproduced in Table G.2-9.

TABLE G.2-9
CONDITIONAL PROBABILITIES OF COLLAPSED APBS FOR INTERNAL PDSS

Collapsed APB Number	PDS 1 Conditional Collapsed APB Probability	PDS 2 Conditional Collapsed APB Probability	PDS 3 Conditional Collapsed APB Probability	PDS 4 Conditional Collapsed APB Probability	PDS 5 Conditional Collapsed APB Probability	PDS 6 Conditional Collapsed APB Probability	PDS 7 Conditional Collapsed APB Probability	PDS 8 Conditional Collapsed APB Probability	PDS 9 Conditional Collapsed APB Probability
1	0	0	0	0	0.053	0.005	0	0.008	0.008
2	0.028	0.028	0	0.024	0.01	0.017	0.011	0.004	0.004
3	0	0	0	0.066	0.503	0.084	0	0.4	0.4
4	0.36	0.36	0.27	0.237	0.11	0.218	0.485	0.163	0.163
5	0	0	0.046	0.005	0.007	0	0	0	0
6	0.074	0.074	0.084	0.063	0.061	0.049	0.012	0.009	0.009
7	0.003	0.003	0.271	0.024	0.084	0	0.308	0.236	0.236
8	0.536	0.536	0.078	0.328	0.088	0.424	0.082	0.08	0.08
9	0	0	0.251	0.253	0.085	0.203	0.074	0.073	0.073
10	0	0	0	0	0	0	0.028	0.028	0.028

The fractional contribution of a given PDS to a given collapsed APB is the product of the PDS frequency and the conditional probability for the collapsed

APB divided by the sum of the products of the PDS frequencies and their conditional probabilities for that same collapsed APB. The following equation describes this relationship:

$$F_{PDS1APB1} = \frac{f_{PDS1} * C_{PDS1APB1}}{(f_{PDS1} * C_{PDS1APB1} + f_{PDS2} * C_{PDS2APB1} \cdots + f_{PDS9} * C_{PDS9APB1}}$$

Where:

$F_{PDS1APB1}$ = fractional contribution of PDS 1 to collapsed APB 1

f_{PDS1} = frequency of PDS 1

$C_{PDS1APB1}$ = conditional probability of collapsed APB 1 for PDS 1

f_{PDS2} = frequency of PDS 2

$C_{PDS2APB1}$ = conditional probability of collapsed APB 1 for PDS 2

f_{PDS9} = frequency of PDS 9

$C_{PDS9APB1}$ = conditional probability of collapsed APB 1 for PDS 9

This is performed for all collapsed APBs. Table G.2-10 summarizes these results.

TABLE G.2-10
FRACTIONAL CONTRIBUTIONS

Collapsed APB Number	Fractional Contribution of PDS 1 to APB	Fractional Contribution of PDS 2 to APB	Fractional Contribution of PDS 3 to APB	Fractional Contribution of PDS 4 to APB	Fractional Contribution of PDS 5 to APB	Fractional Contribution of PDS 6 to APB	Fractional Contribution of PDS 7 to APB	Fractional Contribution of PDS 8 to APB	Fractional Contribution of PDS 9 to APB
1	0.00	0.00	0.00	0.00	8.76×10^{-1}	1.54×10^{-2}	0.00	9.80×10^{-2}	1.05×10^{-2}
2	9.11×10^{-2}	1.09×10^{-1}	0.00	1.03×10^{-1}	4.10×10^{-1}	1.29×10^{-1}	2.37×10^{-2}	1.21×10^{-1}	1.30×10^{-2}
3	0.00	0.00	0.00	8.10×10^{-3}	5.89×10^{-1}	1.83×10^{-2}	0.00	3.47×10^{-1}	3.72×10^{-2}
4	7.19×10^{-2}	8.58×10^{-2}	9.52×10^{-4}	6.25×10^{-2}	2.77×10^{-1}	1.02×10^{-1}	6.40×10^{-2}	3.04×10^{-1}	3.25×10^{-2}
5	0.00	0.00	8.50×10^{-3}	6.90×10^{-2}	9.22×10^{-1}	0.00	0.00	0.00	0.00
6	6.01×10^{-2}	7.17×10^{-2}	1.21×10^{-3}	6.75×10^{-2}	6.24×10^{-1}	9.31×10^{-2}	6.45×10^{-3}	6.82×10^{-2}	7.31×10^{-3}
7	8.01×10^{-4}	9.56×10^{-4}	1.28×10^{-3}	8.46×10^{-3}	2.83×10^{-1}	0.00	5.44×10^{-2}	5.88×10^{-1}	6.30×10^{-2}
8	1.17×10^{-1}	1.39×10^{-1}	3.00×10^{-4}	9.43×10^{-2}	2.41×10^{-1}	2.16×10^{-1}	1.18×10^{-2}	1.63×10^{-1}	1.74×10^{-2}
9	0.00	0.00	1.65×10^{-3}	1.24×10^{-1}	3.98×10^{-1}	1.77×10^{-1}	1.82×10^{-2}	2.54×10^{-1}	2.72×10^{-2}
10	0.00	0.00	0.00	0.00	0.00	0.00	6.02×10^{-2}	8.49×10^{-1}	9.10×10^{-2}

Step 4

The next part of the process is calculating the PDS frequencies based on the collapsed APB frequencies from the PBAPS PSA model (the collapsed APB frequencies used here do not include the dual unit Core damage contribution). This document uses the base case for demonstration purposes; the same process is used for the cases representing SAMA model changes to determine the change in OECR. The PBAPS PSA PDS frequencies are determined by summing the products of the PBAPS PSA collapsed APB frequencies and the fractional contributions of each PDS to the collapsed APBs over all collapsed APBs. The following equation describes this relationship:

$$F_{PDS1PSA} = f_{APB1} * F_{PDS1APB1} + f_{APB2} * F_{PDS1APB2} + f_{APB10} * F_{PDS1APB10}$$

Where:

$f_{PDS1PSA}$ = frequency of PBAPS PSA PDS 1

f_{APB1} = frequency of PBAPS PSA collapsed APB 1

$F_{PDS1APB1}$ = fractional contribution of PDS 1 to collapsed APB 1

f_{APB2} = frequency of PBAPS PSA collapsed APB 2

$F_{PDS1APB2}$ = fractional contribution of PDS 1 to collapsed APB 2

f_{APB10} = frequency of PBAPS PSA collapsed APB 10

$F_{PDS1APB10}$ = fractional contribution of PDS 1 to collapsed APB 10

This process is performed for each PDS. The results are provided in Table G.2-11.

TABLE G.2-11
PDS FREQUENCIES

PDS	PBAPS PSA PDS Frequencies
1	1.04×10^{-7}
2	1.24×10^{-7}
3	3.60×10^{-9}
4	1.91×10^{-7}
5	8.33×10^{-7}
6	2.79×10^{-7}
7	1.06×10^{-7}
8	6.50×10^{-7}
9	6.97×10^{-8}

Step 5

The NUREG/CR-4551 PDS OECR values are determined by multiplying the total OECR (calculated in Step 2) by the fraction of the OECR contributed by the PDS. Table D-1 of NUREG/CR-4551 provides the contribution fractions. Table G.2-12 summarizes the results.

TABLE G.2-12
NUREG/CR-4551 PDS CONTRIBUTIONS TO OECR

PDS	Fractional Contribution of PDS to OECR	NUREG/CR-4551 PDS Contributions to OECR
1	0.02506	6.62×10^{2}
2	0.01819	4.81×10^{2}
3	0.00039	1.03×10^{1}
4	0.01751	4.63×10^{2}
5	0.5701	1.51×10^{4}
6	0.02247	5.94×10^{2}
7	0.02115	5.59×10^{2}
8	0.31504	8.32×10^{3}
9	0.01011	2.67×10^{2}

Steps 6 and 7

These steps provide the PBAPS PSA OECR based on end of license conditions. The PBAPS PSA OECR is calculated by multiplying the NUREG/CR-4551 PDS OECR by the ratio of the PBAPS PSA PDS frequency to the NUREG/CR-4551 PDS frequency. The results are then multiplied by the $P_{34/80}$ ratio to reflect the conditions at the end of the license extension. Table G.2-13 summarizes this process.

TABLE G.2-13
PBAPS PSA OECR

PDS	NUREG/CR-4551 PDS Frequencies	PBAPS PSA PDS Frequencies	Fractional Contribution of PDS to OECR	NUREG/CR-4551 PDS Contributions to OECR	Ratio of PDS Frequencies: PBAPS PSA to NUREG/CR-4551	PBAPS PSA OECR	PBAPS PSA PDS OECR for 2034 Population
1	1.50×10^{-7}	1.04×10^{-7}	0.02506	6.62×10^{2}	6.92×10^{-1}	4.59×10^{2}	1.83×10^{3}
2	1.79×10^{-7}	1.24×10^{-7}	0.01819	4.81×10^{2}	6.92×10^{-1}	3.33×10^{2}	1.33×10^{3}
3	2.65×10^{-9}	3.60×10^{-9}	0.00039	1.03×10^{1}	1.36	1.40×10^{1}	5.59×10^{1}
4	1.98×10^{-7}	1.91×10^{-7}	0.01751	4.63×10^{2}	9.62×10^{-1}	4.45×10^{2}	1.78×10^{3}
5	1.89×10^{-6}	8.33×10^{-7}	0.5701	1.51×10^{4}	4.41×10^{-1}	6.64×10^{3}	2.65×10^{4}
6	3.51×10^{-7}	$2.79\times10^{-}7$	0.02247	5.94×10^{2}	7.95×10^{-1}	4.72×10^{2}	1.89×10^{3}
7	9.92×10^{-8}	1.06×10^{-7}	0.02115	5.59×10^{2}	1.07	5.96×10^{2}	2.38×10^{3}
8	1.40×10^{-6}	6.50×10^{-7}	0.31504	8.32×10^{3}	4.64×10^{-1}	3.87×10^{3}	1.54×10^{4}
9	1.50×10^{-7}	6.97×10^{-8}	0.01011	2.67×10^{2}	6.92×10^{-1}	1.24×10^{2}	4.96×10^{2}
							5.17×10^{4}

The PBAPS PSA OECR based on the assumed conditions at the end of the license extension in 2034 is $51,700.

G.3 COST-BENEFIT ANALYSIS

This sub-section explains how PBAPS calculated the monetary value of the status quo (i.e., accident consequences without SAMA implementation). PBAPS also used this analysis to establish the maximum benefit that a SAMA could achieve if it eliminated all PBAPS risk due to at-power internal events.

The cost-benefit analysis described in this section is performed on a site basis. A single unit is examined in the subsections below and the results are modified to account for the second unit. SAMA implementation costs, which are derived for use in the screening and detailed cost-benefit analyses, are also developed with the understanding that the SAMA would have to be implemented in each unit. The reason for performing the analysis on a site basis is that the implementation costs for modifications that affect both plants will be properly accounted for. For instance, a procedure enhancement is largely applicable to both units and the cost of its development is relevant to the site while installation of a unit specific piece of hardware should be doubled to account for its installation in both units. It is simply a means of maintaining expenditures on the same scale. The Unit 2

PSA model, which has the slightly higher base CDF of the two units, is used in the cost-risk calculations for the site.

The impact of a dual unit core damage scenario was examined as part of this study; however, a detailed Level 3 consequence analysis was not available for a simultaneous release from both units. A PSA sensitivity calculation was performed assuming the consequences of a dual unit core damage event are twice those of a single unit core damage event. Based on a review of the consequences associated with a factor of 2 increase in the source term releases presented in NUREG/CR-4551, this appears to be a conservative assumption. The results of the sensitivity analysis indicate that the consequences of a dual unit core damage event would have to be greater than twice those of a single unit core damage event to have any significant impact on the cost-benefit analysis of the proposed plant changes. Therefore, performance of a detailed dual unit core damage evaluation is not considered to be required as part of the SAMA analysis.

<u>Offsite Exposure Cost</u>

The baseline annual offsite exposure risk was converted to dollars using the NRC's conversion factor of $2,000 per person-rem (Ref. G.8-4, Section 5.7.1.2), and discounting to present value using the NRC standard formula (Ref. G.8-4, Section 5.7.1.3):

$$W_{pha} = C * Z_{pha}$$

Where:

W_{pha} = monetary value of public health risk after discounting

C = $[1-\exp(-rt_f)]/r$

t_f = years remaining until end of facility life = 20 years

r = real discount rate (as fraction) = 0.07/year

Z_{pha} = monetary value of public health risk (accident) per year before discounting ($/year)

The calculated value for C using 20 years and a 7 percent discount rate is 10.76. Therefore, calculating the discounted monetary equivalent of accident risk involves multiplying the dose risk (14.72 person-rem per year) by $2,000 per person-rem and by the C value (10.76). The calculated offsite exposure cost is $316,945.

Offsite Economic Cost Risk

The baseline PBAPS PSA OECR is $51,700. This cost risk is an annual estimate based on the conditions present at the end of the license extension period. The baseline OECR must be discounted to present value as well in order to account for the entire license extension period. This is performed in the same manner as for public health risks and uses the same C value. The resulting estimate is $556,854.

Onsite Exposure Cost Risk

PBAPS evaluated occupational health using the NRC methodology in Ref. G.8-4, Section 5.7.3, which involves separately evaluating “immediate” and long-term doses.

Immediate Dose - For the case where the plant is in operation, the equations that NRC recommends using (Ref. G.8-4, Sections 5.7.3 and 5.7.3.3) is:

Equation 1:

$$W_{IO} = R\{(FD_{IO})_S - (FD_{IO})_A\} * \left\{\frac{[1 - \exp(-rt_f)]}{r}\right\}$$

Where:

W_{IO} = monetary value of accident risk avoided due to immediate doses, after discounting

R = monetary equivalent of unit dose ($/person-rem)

F = accident frequency (events/yr)

D_{IO} = immediate occupational dose (person-rem/event)

S = subscript denoting status quo (current conditions)

A = superscript denoting after implementation of proposed action

r = real discount rate

t_f = years remaining until end of facility life.

The values used in the PBAPS analysis are:

R = \$2,000/person-rem

r = 0.07/year

D_{IO} = 3,300 person-rem/accident (best estimate, from Ref. G.8.4, Section 5.7.3.1)

t_f = 20 years (license extension period)

F = 4.5E-6 (baseline CDF) events/year

For the basis discount rate, assuming F_A is zero, the best estimate of the immediate dose cost is:

$$W_{10} = R(FD_{IO})_S * \left\{\frac{[1 - \exp(-rt_f)]}{r}\right\}$$

$$= \quad 2000 * (4.5E-6 * 3{,}300) * \left\{\frac{[1 - \exp(-0.07 * 20)]}{0.07}\right\}$$

$$= \quad \$322$$

Long-Term Dose - For the case where the plant is in operation, the NRC equation (Ref. G.8-4, Sections 5.7.3 and 5.7.3.3) is:

Equation 2:

$$W_{LTO} = R\{(FD_{LTO})_S - (FD_{LTO})_A\} * \left\{\frac{[1 - \exp(-rt_f)]}{r}\right\} * \left\{\frac{[1 - \exp(-rm)]}{rm}\right\}$$

Where:

W_{IO} = monetary value of accident risk avoided long-term doses, after discounting, \$

m = years over which long-term doses accrue

The values used in the PBAPS analysis are:

R = \$2,000/person-rem

r = 0.07/year

D_{LTO} = 20,000 person-rem/accident (best estimate, Ref. G.8-4, Section 5.7.3.1)

m = 10 years (estimate)

t_f = 20 years (license extension period)

F = 4.5E-6 (baseline CDF) events/year

For the basis discount rate, assuming F_A is zero, the best estimate of the long-term dose is:

$$W_{LTO} = R\,(FD_{LTO})_S * \left\{\frac{[1-\exp(-rt_f)]}{r}\right\} * \left\{\frac{[1-\exp(-rm)]}{rm}\right\}$$

$$= \quad 2000 * (4.5E-6 * 20{,}000) * \left\{\frac{[1-\exp(-0.07*20)]}{0.07}\right\} * \left\{\frac{[1-\exp(-0.07*10)]}{0.07*10}\right\}$$

$$= \quad \$1{,}403$$

Total Occupational Exposure - Combining Equations 1 and 2 above and using the above numerical values, the total accident related on-site (occupational) exposure avoided (W_O) based one unit's contribution to independent, single unit core damage is:

$$W_O = W_{IO} + W_{LTO} = (\$322 + \$1{,}403) = \$1{,}725$$

Onsite Cleanup and Decontamination Cost

The net present value that NRC provides for cleanup and decontamination for a single event is \$1.1 billion, discounted over a 10-year cleanup period

(Ref. G.8-4, Section 5.7.6.1). NRC uses the following equation in integrating the net present value over the average number of remaining service years:

$$U_{CD} = \left[\frac{PV_{CD}}{r}\right][1 - \exp(-rt_f)]$$

Where:

U_{CD} = Net present value of cost of cleanup and decontamination over the life of the facility

PV_{CD} = Net present value of a single event

r = real discount rate

t_f = years remaining until end of facility life.

The values used in the PBAPS analysis are:

PV_{CD} = \$1.1E9

r = 0.07/year

t_f = 20 years

The resulting net present value of cleanup integrated over the license renewal term, \$1.18E10 must be multiplied by the baseline CDF of 4.5E-6 to determine the expected value of cleanup and decontamination costs. The resulting monetary equivalent is \$53,643.

Replacement Power Cost

Long-term replacement power costs was determined following the NRC methodology in Ref. G.8-4, Section 5.7.6.2. The net present value of replacement power for a single event, PV_{RP}, was determined using the following equation:

$$PV_{RP} = \left[\frac{\$1.2E8}{r}\right] * [1 - \exp(-rt_f)]^2$$

Where:

PV_{RP} = net present value of replacement power for a single event, ($)

r = 0.07/year

t_f = 20 years (license renewal period)

To attain a summation of the single-event costs over the entire license renewal period, the following equation is used:

$$U_{RP} = \left[\frac{PV_{RP}}{r}\right] * [1 - \exp(-rt_f)]^2$$

Where:

U_{RP} = net present value of replacement power over life of facility ($-year)

After applying a correction factor to account for PBAPS size relative to the "generic" reactor described in NUREG/BR-0184 (i.e., 1159 MWe/910 MWe) and multiplying by 2 to account for the assumption that the remaining unit has to shut down after a core damage event, the replacement power costs are determined to be $2.01×10 ($-year). Multiplying this value by the baseline CDF (4.5×10^{-6}) results in a replacement power cost of $91,067.

<u>Baseline Screening</u>

The sum of the baseline costs for a single unit core damage event is as follows:

Offsite exposure cost	=	$316,945
Offsite economic cost	=	$556,854
Onsite exposure cost	=	$1,725
Onsite cleanup cost	=	$53,643
Replacement Power cost	=	$91,067
Total cost	=	$1,020,234

To account for the contribution from both units, this answer is multiplied by 2 to yield $2,040,468.

This combined cost estimate for both Peach Bottom units was used in screening out SAMAs that are not economically feasible; if the estimated cost of implementing a SAMA exceeded $2.04 million, it was discarded from further analysis. Exceeding this threshold would mean that a SAMA would not have a positive net value even if it could eliminate all severe accident costs. On the other hand, if the cost of implementation is less than this value, then a more detailed examination of the potential fractional risk benefit that can be attributed to the SAMA is performed.

G.4 PHASE I SAMA ANALYSIS: SAMA CANDIDATES AND SCREENING PROCESS

An initial list of 207 SAMA candidates was developed from lists of Severe Accident Mitigation Alternatives at other nuclear power plants (Refs. G.8-6, G.8-10, G.8-11, G.8-13, G.8-15, G.8-18, and G.8-19), NRC documents (Refs. G.8-5, G.8-8, G.8-9, G.8-12, G.8-14, G.8-21, and G.8-22), and documents related to advanced power reactor designs (ABWR SAMAs) (Refs. G.8-7, G.8-16, and G.8-17). Table G.4-1 provides this list. This initial list was then screened to remove those that were not applicable to Peach Bottom due to design differences. The SAMA screening process is summarized in Figure G.4-1.

TABLE G.4-1
PHASE I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
1	Cap downstream piping of normally closed component cooling water drain and vent valves.	1	SAMA would reduce the frequency of a loss of component cooling event, a large portion of which was derived from catastrophic failure of one of the many single isolation valves.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A
2	Enhance loss of component cooling procedure to facilitate stopping reactor coolant pumps.	2	SAMA would reduce the potential for reactor coolant pump (RCP) seal damage due to pump bearing failure.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A
3	Enhance loss of component cooling procedure to present desirability of cooling down reactor coolant system (RCS) prior to seal LOCA.	2	SAMA would reduce the potential for RCP seal failure.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
4	Provide additional training on the loss of component cooling.	2	SAMA would potentially improve the success rate of operator actions after a loss of component cooling (to restore RCP seal damage).	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A
5	Provide hardware connections to allow another essential raw cooling water system to cool charging pump seals.	1 2	SAMA would reduce effect of loss of component cooling by providing a means to maintain the centrifugal charging pump seal injection after a loss of component cooling.	#1 - N/A	PWR issue. BWRs do not have charging pumps and seal LOCAs for other BWR pumps are not significant contributors to plant risk.	NUREG-1560	N/A
5A	Procedure changes to allow cross connection of motor cooling for RHRSW pumps.	12	SAMA would allow continued operation of both RHRSW pumps on a failure of one train of PSW.	#1 - N/A	The equivalent system at PBAPS to RHRSW is HPSW. HPSW does not depend on any other systems for cooling. HPSW takes suction directly from the Ultimate Heat Sink and the pump motors are self cooled.	PBAPS PRA	N/A
6	Proceduralize shedding component cooling water loads to extend component cooling heatup on loss of essential raw cooling water.	2	SAMA would increase time before the loss of component cooling (and reactor coolant pump seal failure) in the loss of essential raw cooling water sequences.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
7	Increase charging pump lube oil capacity.	2	SAMA would lengthen the time before centrifugal charging pump failure due to lube oil.	#1 - N/A	PWR issue. BWRs do not have charging pumps and the potential equivalents, the CRD pumps, are not risk significant components.	NUREG-1560	N/A
8	Eliminate the RCP thermal barrier dependence on component cooling such that loss of component cooling does not result directly in core damage.	2	SAMA would prevent the loss of recirculation pump seal integrity after a loss of component cooling. Watts Bar Nuclear Plant IPE said that they could do this with essential raw cooling water connection to charging pump seals.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A
9	Add redundant DC control power for PSW pumps C & D.	3	SAMA would increase reliability of PSW and decrease core damage frequency due to a loss of SW.	#1 - N/A	The equivalent system at PBAPS is the NSW. No NSW system dependencies on plant internal DC are identified in the PRA. The NSW depends on offsite AC only.	PBAPS PRA	N/A
10	Create an independent RCP seal injection system, with a dedicated diesel.	1	SAMA would add redundancy to RCP seal cooling alternatives, reducing CDF from loss of component cooling or service water or from a station blackout event.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
11	Use existing hydro-test pump for RCP seal injection.	4	SAMA would provide an independent seal injection source, without the cost of a new system.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs that do not rely on isolation condensers.	NUREG-1560	N/A
12	Replace ECCS pump motor with air-cooled motors.	1 14	SAMA would eliminate ECCS dependency on component cooling system (but not on room cooling).	#1 - N/A	PBAPS has evaluated this before and determined that this SAMA is not required.	Table 3.4-2 in Evaluation of Peach Bottom Accident Management Insights with Regards to BWROG EPG/SAG Strategies	N/A
13	Install improved RCS pumps seals.	1	SAMA would reduce probability of RCP seal LOCA by installing RCP seal O-ring constructed of improved materials	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.	NUREG-1560	N/A
14	Install additional component cooling water pump.	1	SAMA would reduce probability of loss of component cooling leading to RCP seal LOCA.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.	NUREG-1560	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
15	Prevent centrifugal charging pump flow diversion from the relief valves.	1	SAMA modification would reduce the frequency of the loss of RCP seal cooling if relief valve opening causes a flow diversion large enough to prevent RCP seal injection.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.	NUREG-1560	N/A
16	Change procedures to isolate RCP seal letdown flow on loss of component cooling, and guidance on loss of injection during seal LOCA.	1	SAMA would reduce CDF from loss of seal cooling.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.	NUREG-1560	N/A
17	Implement procedures to stagger high-pressure safety injection (HPSI) pump use after a loss of service water.	1	SAMA would allow HPSI to be extended after a loss of service water.	#1 - N/A	The approximate equivalent to HPSI in a BWR are the HPCI and RCIC systems; these do not directly depend on NSW/ESW/ECW cooling. Room cooling is provided by these service water systems, but RCIC and HPCI can operate without room cooling. Therefore, staggering their operation is not required.	1) PBAPS PRA 2) DBD No. P-T-13, Rev. 5, p. 57 3) SE-11 Bases, Rev. 11, p. 13	N/A
18	Use fire protection system pumps as a backup seal injection and high-pressure makeup.	1	SAMA would reduce the frequency of the RCP seal LOCA and the SBO CDF.	#1 - N/A	PWR RCP seal leakage issue. Although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to CDF in BWRs.	NUREG-1560	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
19	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	1 14	SAMA would reduce the frequency of the loss of component cooling water and service water.	Retain			1
20	Procedure enhancements and operator training in support system failure sequences, with emphasis on anticipating problems and coping.	1 2 14	SAMA would potentially improve the success rate of operator actions subsequent to support system failures.	#2 - Similar item is addressed under other proposed SAMAs.	See 19, 24, 54, 60, 61, 62, 67, 108		N/A
21	Improved ability to cool the residual heat removal heat exchangers.	1	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.	Retain			2
22	Provide reliable power to control building fans.	2	SAMA would increase availability of control room ventilation on a loss of power.	#3 - Already installed.	The CR HVAC system is designed with redundant active components and redundant Class 1E power supplies for the CR Fresh Air Supply System and the CR Emergency Ventilation Filter System.	DBD No. P-S-08B, Rev. 8	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
23	Provide a redundant train of ventilation.	1	SAMA would increase the availability of components dependent on room cooling.	#3 – Already installed	It has been determined that room cooling is not required for successful operation of RHR, LPCS, HPCI or RCIC at PBAPS (HPCI and RCIC are modeled such that failure of the gland seal condensers is required before room cooling is considered as a necessary support function). The only system with a true room cooling dependency at PBAPS is the Emergency AC power system. The EDG rooms require room cooling for success, but these rooms are already equipped with redundant fan trains.		N/A
24	Procedures for actions on loss of HVAC.	12 14	SAMA would provide for improved credit to be taken for loss of HVAC sequences (improved affected electrical equipment reliability upon a loss of control building HVAC).	#3 - Already installed.	1) No loss of HVAC initiating events are identified for PBAPS. 2) Loss of HVAC due to SBO is addressed. 3) Placing control room emergency ventilation in service is proceduralized.	1) PBAPS PRA 2) SE-11 procedure 3) SO 40D.7.B	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
25	Add a diesel building switchgear room high temperature alarm.	1 14	SAMA would improve diagnosis of a loss of switchgear room HVAC. Option 1: Install high temp alarm. Option 2: Redundant louver and thermostat	#1 - N/A	Diesel Generator ventilation supply fans start upon a diesel start and supply combustion air as well as ventilation for diesel support equipment within the diesel room. Electrical distribution equipment associated with diesel support equipment is located in open areas of the reactor building and is not subject to failure on loss of ventilation.	DBD No. P-S-07, Rev 12, p. 39	N/A
26	Create ability to switch fan power supply to DC in an SBO event.	1	SAMA would allow continued operation in an SBO event. This SAMA was created for reactor core isolation cooling system room at Fitzpatrick Nuclear Power Plant.	#1 - N/A	Equipment in the RCIC pump room has demonstrated operability for room temp up to 163F for 12 hrs. In SBO, 163F is not reached at 4 hrs. At 8 hrs, 163F is barely exceeded. Room cooling therefore not required during the mission time of RCIC.	DBD No. P-T-13, Rev. 5, p57	N/A
27	Delay containment spray actuation after large LOCA.	2 14	SAMA would lengthen time of RWST availability.	#1 - N/A	The RHR containment spray modes take suction from the suppression pool. The RWST volume is therefore not affected by containment spray. Capability exists to transfer water from the other unit's CST.	PBAPS PRA Procedures SE-11, SAMP-1, SAMP-2	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
28	Install containment spray pump header automatic throttle valves.	4 8	SAMA would extend the time over which water remains in the RWST, when full CS flow is not needed	#2 - Similar item is addressed under other proposed SAMAs.	See 27		N/A
29	Install an independent method of suppression pool cooling.	5 6	SAMA would decrease the probability of loss of containment heat removal.	Retain			3
30	Develop an enhanced drywell spray system.	5 6 14	SAMA would provide a redundant source of water to the containment to control containment pressure, when used in conjunction with containment heat removal.	#3 - Already installed.	The HPSW system take suction from the Conowingo Pond and can discharge to the RPV or containment sprays via the RHR system.	PBAPS PRA Procedures T-245, T-205	N/A
31	Provide dedicated existing drywell spray system.	5 6	SAMA would provide a source of water to the containment to control containment pressure, when used in conjunction with containment heat removal. This would use an existing spray loop instead of developing a new spray system.	#3 - Already installed.	The drywell spray function is integral to the RHR system. Procedure T-204-2 provides instructions for manual initiation of the Containment Spray Mode of RHR.	PBAPS PRA. Procedure T-204-2	N/A
32	Install an unfiltered hardened containment vent.	5 6 14	SAMA would provide an alternate decay heat removal method for non-ATWS events, with the released fission products not being scrubbed.	#3 - Already installed.	The hardened (pipe) vent, added to comply with Generic Letter 89-16, is installed between Torus valves AO-7C-2511 and AO-7C-2512, and includes a rupture disc (set at 30 psig).	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
33	Install a filtered containment vent to remove decay heat.	5 6	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	Retain	1) Hardened vent is filtered via the SP	2) A filter-like system could be added	4
34	Install a containment vent large enough to remove ATWS decay heat.	5 6	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	Retain	Add large vent capability		5
35	Create/enhance hydrogen recombiners with independent power supply.	5 11	SAMA would reduce hydrogen detonation at lower cost, Use either 1) a new independent power supply 2) a nonsafety-grade portable generator 3) existing station batteries 4) existing AC/DC independent power supplies.	#1 - N/A	The PBAPS primary containment is inert. The CAD system is designed to control the O2 and H2 concentrations by venting and purging with nitrogen. Hydrogen recombiners have limited capability for conditions with high hydrogen.	PBAPS Level 2 PRA	N/A
35A	Install hydrogen recombiners.	11	SAMA would provide a means to reduce the chance of hydrogen detonation.	#1 - N/A	The PBAPS primary containment is inert. The CAD system is designed to control the O2 and H2 concentrations by venting and purging with nitrogen. Hydrogen recombiners have limited capability for conditions with high hydrogen.	PBAPS Level 2 PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
36	Create a passive design hydrogen ignition system.	4	SAMA would reduce hydrogen denotation system without requiring electric power.	#1 - N/A	The PBAPS primary containment is inert. The CAD system is designed to control the O2 and H2 concentrations by venting and purging with nitrogen. Hydrogen recombiners have limited capability for conditions with high hydrogen.	PBAPS Level 2 PRA	N/A
37	Create a large concrete crucible with heat removal potential under the basemat to contain molten core debris.	5 6	SAMA would ensure that molten core debris escaping from the vessel would be contained within the crucible. The water cooling mechanism would cool the molten core, preventing a melt-through of the basemat.	#5 - Cost would be more than risk benefit	Core retention devices have been investigated in previous studies. IDCOR concluded that "core retention devices are not effective risk reduction devices for degraded core events". Other evaluations have shown the worth value for a core retention device to be on the order of $7000 (averted cost-risk) compared to an estimated implementation cost of over $1 million (per unit).	Supplement 2 to NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, December 1999 for Oconee Nuclear Station, and IDCOR Technical Summary Report, November 1984	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
38	Create a water-cooled rubble bed on the pedestal.	5 6	SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.	#5 - Cost would be more than risk benefit	Core retention devices have been investigated in previous studies. IDCOR concluded that "core retention devices are not effective risk reduction devices for degraded core events". Other evaluations have shown the worth value for a core retention device to be on the order of $7000 (averted cost-risk) compared to an estimated implementation cost of over $1 million (per unit).	Supplement 2 to NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, December 1999 for Oconee Nuclear Station, and IDCOR Technical Summary Report, November 1984	N/A
39	Provide modification for flooding the drywell head.	5 6	SAMA would help mitigate accidents that result in the leakage through the drywell head seal.	#1 - N/A	BWR Mark I risk is typically dominated by events that result in early failure of the drywell shell due to direct contact with core debris and events that bypass the containment. This is also true at Peach Bottom. The head flooding system would, therefore, not be expected to have any significant impact on the overall risk.	Results of Mark I plant IPEs and NUREG-1150	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
40	Enhance fire protection system and/or standby gas treatment system hardware and procedures.	6	SAMA would improve fission product scrubbing in severe accidents.	#1 - N/A	Current Fire Protection and Standby Gas Treatment Systems do not have sufficient capacity to handle the loads from severe accidents that result in a bypass or breach of the containment. Loads produced as a result of RPV or containment blowdown would require large filtering capacities. These filtered vented systems have been previously investigated and found not to provide sufficient cost benefit.	IDCOR Technical Summary Report, November 1984	N/A
41	Create a reactor cavity flooding system.	1 3 7 8 14	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#3 - Already installed.	Flooding of the PBAPS containment (incl. reactor cavity) is proceduralized in the EOPs. In addition to the normal injection sources, HPSW, Condensate Transfer, Refueling Water Transfer, Fire and SBLC can be used.	Alternate Injection Sources PBAPS Level II PRA System Notebook	N/A
42	Create other options for reactor cavity flooding.	1 14	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#3 - Already installed.	Flooding of the PBAPS containment (incl. reactor cavity) is proceduralized in the EOPs. In addition to the normal injection sources, HPSW, Condensate Transfer, Refueling Water Transfer, Fire and SBLC can be used.	Alternate Injection Sources PBAPS Level II PRA System Notebook	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
43	Enhance air return fans (ice condenser plants).	1	SAMA would provide an independent power supply for the air return fans, reducing containment failure in SBO sequences.	#1 - N/A	PBAPS is not an ice-condenser plant	PBAPS PRA	N/A
44	Create a core melt source reduction system.	9	SAMA would provide cooling and containment of molten core debris. Refractory material would be placed underneath the reactor vessel such that a molten core falling on the material would melt and combine with the material. Subsequent spreading and heat removal form the vitrified compound would be facilitated, and concrete attack would not occur	#5 - Cost would be more than risk benefit	Core retention devices have been investigated in previous studies. IDCOR concluded that "core retention devices are not effective risk reduction devices for degraded core events". Other evaluations have shown the worth value for a core retention device to be on the order of $7000 compared to an estimated implementation cost of over $1 million.	Supplement 2 to NUREG-1437, Generic Environmental Impact Statement for License Renewal of Nuclear Plants, December 1999 for Oconee Nuclear Station, and IDCOR Technical Summary Report, November 1984	N/A
45	Provide a containment inerting capability.	7 8	SAMA would prevent combustion of hydrogen and carbon monoxide gases.	#3 - Already installed.	Containment is inerted with nitrogen during normal operation. CAD system also available.	PBAPS Level 2 PRA	N/A
46	Use the fire protection system as a backup source for the containment spray system.	4	SAMA would provide redundant containment spray function without the cost of installing a new system.	Retain			6

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
47	Install a secondary containment filter vent.	10	SAMA would filter fission products released from primary containment.	#3 - Already installed.	Standby Gas Treatment System inlet can connect the reactor building refueling floor ventilation exhaust duct.	PBAPS Level 2 PRA	N/A
48	Install a passive containment spray system.	10	SAMA would provide redundant containment spray method without high cost.	Retain			7
49	Strengthen primary/secondary containment.	10 11	SAMA would reduce the probability of containment overpressurization to failure.	#5 - Cost would be more than risk benefit	BWR Mark I risk is typically dominated by events that result in early failure of the drywell shell due to direct contact with core debris and events that bypass the containment. Strengthening the primary /secondary containment would have a small impact on the overall risk of these accidents. In addition, the estimated implementation cost would be over $1 million/site.	Results of Mark I plant IPEs and NUREG-1150	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
50	Increase the depth of the concrete basemat or use an alternative concrete material to ensure melt-through does not occur.	11	SAMA would prevent basemat melt-through.	#5 - Cost would be more than risk benefit	Core retention devices have been investigated in previous studies. IDCOR concluded that "core retention devices are not effective risk reduction devices for degraded core events". Other evaluations have shown the worth value for a core retention device to be on the order of $7000 compared to an estimated implementation cost of over $1 million/site.	Supplement 2 to NUREG-1437, Generic Environmental Impact Statement for License renewal of Nuclear Plants, December 1999 for Oconee Nuclear Station, and IDCOR Technical Summary Report, November 1984	N/A
51	Provide a reactor vessel exterior cooling system.	11	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	#2 - Similar item is addressed under other proposed SAMAs.	See 41		N/A
52	Construct a building to be connected to primary/secondary containment that is maintained at a vacuum.	11	SAMA would provide a method to depressurize containment and reduce fission product release.	Retain			8
53	Not used.	N/A	N/A	N/A	#N/A	N/A	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
54	Proceduralize alignment of spare diesel to shutdown board after loss of offsite power and failure of the diesel normally supplying it.	1 3 7	SAMA would reduce the SBO frequency.	Retain	Install spare D/G (See 56)		9
55	Not used.	N/A	N/A	N/A	#N/A	N/A	N/A
56	Provide an additional diesel generator.	1 3 7 11 14	SAMA would increase the reliability and availability of onsite emergency AC power sources.	Retain	Install spare D/G		10
57	Provide additional DC battery capacity.	1 3 7 11 12	SAMA would ensure longer battery capability during an SBO, reducing the frequency of long-term SBO sequences.	Retain	Providing additional DC battery capacity could extend HPCI/RCIC operability and allow more credit for AC power recovery. This would decrease the frequency of core damage and offsite releases.	PBAPS PRA	11
58	Use fuel cells instead of lead-acid batteries.	11	SAMA would extend DC power availability in an SBO.	Retain			12
59	Procedure to cross-tie high-pressure core spray diesel.	1	SAMA would improve core injection availability by providing a more reliable power supply for the high-pressure core spray pumps.	#1 - N/A	PBAPS does not have a high-pressure core spray system. The HPCI (equivalent system) is turbine driven.	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
60	Improve 4.16-kV bus cross-tie ability.	1 14	SAMA would improve AC power reliability.	#3 - Already installed.	Enhancements were made to procedure SE-11 to cross-tie 4kV buses that consider all permutations of diesel generators failures.	SE-11 Evaluation of Peach Bottom Accident Management Insights with Regards to BWROG EPG/SAG Strategies	N/A
61	Incorporate an alternate battery charging capability.	1 8 9 14	SAMA would improve DC power reliability by either cross-tying the AC busses, or installing a portable diesel-driven battery charger.	#3 - Already installed.	Cross-tying of electrical buses, allowing chargers to be supplied from other divisions are proceduralized. Specific direction is given to supply power to all battery chargers. Procedural and hardware enhancements maybe pursued to allow use of portable battery chargers, but is not crucial considering the extensive cross-tie capability.	SE-11	N/A
62	Increase/improve DC bus load shedding.	1 8 14	SAMA would extend battery life in an SBO event.	#3 - Already installed.	Plant DC load shedding procedures have been enhanced to increase the probability of successful load shed during SBO conditions.		N/A
63	Replace existing batteries with more reliable ones.	11 14	SAMA would improve DC power reliability and thus increase available SBO recovery time.	#3 - Already installed.	Reliable batteries are already installed.	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
63A	Mod for DC Bus A reliability.	1	SAMA would increase the reliability of AC power and injection capability. Loss of DC Bus A causes a loss of main condenser, prevents transfer from the main transformer to offsite power, and defeats one half of the low vessel pressure permissive for LPCI/CS injection valves.	#1 - N/A	PBAPS Unit 2 has 4 125V DC and 2 250V DC buses. No loss of a single DC bus leads to loss of condenser. Transfer from main transformer to offsite power also not affected.	PBAPS PRA	N/A
64	Create AC power cross-tie capability with other unit.	1 8 9 14	SAMA would improve AC power reliability.	#3 - Already installed.	Procedure SE-11 describes cross-tying 4 kV buses to feed equipment from various 4 kV buses with other diesel generators if the normal diesel generator(s) fails		N/A
65	Create a cross-tie for diesel fuel oil.	1	SAMA would increase diesel fuel oil supply and thus diesel generator, reliability.	#3 - Already installed.	Each of the 4 diesel fuel oil storage tanks can be supplied from 2 other diesel fuel storage tanks.	Procedure AO 52D-1, Rev. 5	N/A
66	Develop procedures to repair or replace failed 4-kV breakers.	1	SAMA would offer a recovery path from a failure of the breakers that perform transfer of 4.16-kV non-emergency busses from unit station service transformers, leading to loss of emergency AC power.	Retain			13

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
67	Emphasize steps in recovery of offsite power after an SBO.	1 14	SAMA would reduce human error probability during offsite power recovery.	#3 - Already implemented.	Restoring power from offsite sources after SBO is proceduralized. Numerous procedural enhancements have been implemented for offsite AC power recovery and to cross-tie AC busses.	SO 53.7.G AO 50F.2-2(3) SE-11 Attachment Z	N/A
68	Develop a severe weather conditions procedure.	1 13	For plants that do not already have one, this SAMA would reduce the CDF for external weather-related events.	#3 - Already implemented.	PREPARATION FOR SEVERE WEATHER guideline provides the station with items to be considered in the event severe weather is forecasted to impact Peach Bottom.	Procedure AG-108, Rev. 4	N/A
69	Develop procedures for replenishing diesel fuel oil.	1	SAMA would allow for long-term diesel operation.	#3 - Already implemented.	Instructions are provided to fill a Diesel Fuel Oil Storage Tank from a fuel oil delivery truck.	SO 52D.3.A	N/A
70	Install gas turbine generator.	1 14	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	Retain			14
71	Not used.	N/A	N/A	N/A	#N/A	N/A	N/A
72	Create a backup source for diesel cooling. (Not from existing system)	1	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	#3 - Already installed.	The ECW pump provides back-up to the ESW system that cools the diesel generators. Each pump (ESW A, ESW B, and the ECW pump are 100% capacity pumps).	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
73	Use fire protection system as a backup source for diesel cooling.	1	This SAMA would provide a redundant and diverse source of cooling for the diesel generators, which would contribute to enhanced diesel reliability.	#2 - Similar item is addressed under other proposed SAMAs.	See 72		N/A
74	Provide a connection to an alternate source of offsite power.	1	SAMA would reduce the probability of a loss of offsite power event.	#3 - Already installed.	The Station Blackout line from Conowingo can provide power to all eight 4 kV buses for the various station blackout scenarios.	PBAPS PRA	N/A
75	Bury offsite power lines.	1	SAMA could improve offsite power reliability, particularly during severe weather.	#3 - Already installed.	The Conowingo tie-line is buried under the river bed from the dam's switchyard to the transformer on the PBAPS site.	DBD No. P-T-13, Revision 6, p. 43	N/A
76	Replace anchor bolts on diesel generator oil cooler.	1	Millstone Nuclear Power Station found a high seismic SBO risk due to failure of the diesel oil cooler anchor bolts. For plants with a similar problem, this would reduce seismic risk. Note that these were Fairbanks Morse DGs.	#3 - Already installed.	DGs are Colt Industries Units. An A-46 anchorage evaluation was performed which demonstrated that the anchorage was acceptable.	PBAPS IPEEE	N/A
77	Change undervoltage (UV), auxiliary feedwater actuation signal (AFAS) block and high pressurizer pressure actuation signals to 3-out-of-4, instead of 2-out-of-4 logic.	1	SAMA would reduce risk of 2/4 inverter failure.	#1 - N/A	PWR issue. N/A to BWR		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
78	Provide DC power to the 120/240-V vital AC system from the Class 1E station service battery system instead of its own battery.	12	SAMA would increase the reliability of the 120-VAC Bus.	#4 - No significant safety benefit	1) Loss of 120V AC is not an Initiating Event 2) 120 VAC is not a risk significant support system	PBAPS PRA	N/A
79	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture (SGTR).	1	SAMA would enhance depressurization during a SGTR.	#1 - N/A	PWR issue. N/A to BWR		N/A
80	Improve SGTR coping abilities.	1 4 11	SAMA would improve instrumentation to detect SGTR, or additional system to scrub fission product releases.	#1 - N/A	PWR issue. N/A to BWR		N/A
81	Add other SGTR coping abilities.	4 10 11	SAMA would decrease the consequences of an SGTR.	#1 - N/A	PWR issue. N/A to BWR		N/A
82	Increase secondary side pressure capacity such that an SGTR would not cause the relief valves to lift.	10 11	SAMA would eliminate direct release pathway for SGTR sequences.	#1 - N/A	PWR issue. N/A to BWR		N/A
83	Replace steam generators (SG) with a new design.	1	SAMA would lower the frequency of an SGTR.	#1 - N/A	PWR issue. N/A to BWR		N/A
84	Revise emergency operating procedures to direct that a faulted SG be isolated.	1	SAMA would reduce the consequences of an SGTR.	#1 - N/A	PWR issue. N/A to BWR		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
85	Direct SG flooding after a SGTR, prior to core damage.	10	SAMA would provide for improved scrubbing of SGTR releases.	#1 - N/A	PWR issue. N/A to BWR		N/A
86	Implement a maintenance practice that inspects 100% of the tubes in a SG.	11	SAMA would reduce the potential for an SGTR.	#1 - N/A	PWR issue. N/A to BWR		N/A
87	Locate residual heat removal (RHR) inside of containment.	10	SAMA would prevent intersystem LOCA (ISLOCA) out the RHR pathway.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
88	Not used.	N/A	N/A	N/A	#N/A	N/A	N/A
89	Install additional instrumentation for ISLOCAs.	3 4 7 8	SAMA would decrease ISLOCA frequency by installing pressure of leak monitoring instruments in between the first two pressure isolation valves on low-pressure inject lines, RHR suction lines, and HPSI lines.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
90	Increase frequency for valve leak testing.	1	SAMA could reduce ISLOCA frequency.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
91	Improve operator training on ISLOCA coping.	1	SAMA would decrease ISLOCA effects.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
92	Install relief valves in the CC System.	1	SAMA would relieve pressure buildup from an RCP thermal barrier tube rupture, preventing an ISLOCA.	#1 - N/A	PWR issue. N/A to BWR	IN-92-36, and its additional supplement	N/A
93	Provide leak testing of valves in ISLOCA paths.	1	SAMA would help reduce ISLOCA frequency. At Kewaunee Nuclear Power Plant, four MOVs isolating RHR from the RCS were not leak tested.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
94	Revise EOPs to improve ISLOCA identification.	1	SAMA would ensure LOCA outside containment could be identified as such. Salem Nuclear Power Plant had a scenario where an RHR ISLOCA could direct initial leakage back to the pressurizer relief tank, giving indication that the LOCA was inside containment.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
95	Ensure all ISLOCA releases are scrubbed.	1	SAMA would scrub all ISLOCA releases. One example is to plug drains in the break area so that the break point would cover with water.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
96	Add redundant and diverse limit switches to each containment isolation valve.	1	SAMA could reduce the frequency of containment isolation failure and ISLOCAs through enhanced isolation valve position indication.	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A
97	Modify swing direction of doors separating turbine building basement from areas containing safeguards equipment.	1	SAMA would prevent flood propagation, for a plant where internal flooding from turbine building to safeguards areas is a concern.	#4 - No significant safety benefit	Flooding from Turbine Hall into adjacent buildings considered to have negligible impact.	PBAPS Internal Flooding Analysis in PRA	N/A
98	Improve inspection of rubber expansion joints on main condenser.	1 14	SAMA would reduce the frequency of internal flooding, for a plant where internal flooding due to a failure of circulating water system expansion joints is a concern.	#4 - No significant safety benefit	PBAPS has evaluated this before and determined that no additional action would be beneficial in reducing the frequency.	Evaluation of Peach Bottom Accident Management Insights with Regards to BWROG EPG/SAG Strategies	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
99	Implement internal flood prevention and mitigation enhancements.	1	This SAMA would reduce the consequences of internal flooding.	#4 - No significant safety benefit	The total core damage frequency attributable to internal flooding for each Unit is 9E-08 per year. PBAPS is extremely flood resistant for all safety related and ECCS equipment, as shown by the extremely low core damage frequencies	PBAPS Internal Flooding Analysis in PRA	N/A
100	Implement internal flooding improvements such as those implemented at Fort Calhoun.	1	This SAMA would reduce flooding risk by preventing or mitigating rupture in the RCP seal cooler of the component cooling systeman ISLOCA in a shutdown cooling line, an auxiliary feedwater (AFW) flood involving the need to remove a watertight door.	#1 - N/A	PWR issue. N/A to BWR		N/A
101	Install a digital feedwater upgrade.	1	This SAMA would reduce the chance of a loss of main feedwater following a plant trip.	#3 - Already installed.	Already installed at Peach Bottom.	PBAPS PRA Section 5	N/A
102	Perform surveillances on manual valves used for backup AFW pump suction.	1	This SAMA would improve success probability for providing alternative water supply to the AFW pumps.	#1 - N/A	PWR issue. N/A to BWR		N/A
103	Install manual isolation valves around AFW turbine-driven steam admission valves.	1	This SAMA would reduce the dual turbine-driven AFW pump maintenance unavailability.	#1 - N/A	PWR issue. N/A to BWR		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
104	Install accumulators for turbine-driven AFW pump flow control valves (CVs).	4 8	This SAMA would provide control air accumulators for the turbine-driven AFW flow CVs, the motor-driven AFW pressure CVs and SG power-operated relief valves (PORVs). This would eliminate the need for local manual action to align nitrogen bottles for control air during a LOOP.	#1 - N/A	PWR issue. N/A to BWR		N/A
105	Proceduralize intermittent operation of HPCI.	1	SAMA would allow for extended duration of HPCI availability.	Retain	HPCI can normally be shut down within 10 minutes after a LOOP and reactor scram, if RCIC can maintain level.	SE-11 BASES Rev.11 p.13	15
106	Increase the reliability of safety relief valves by adding signals to open them automatically.	12	SAMA reduces the probability of a certain type of medium break LOCA. Hatch evaluated medium LOCA initiated by an MSIV closure transient with a failure of SRVs to open. Reducing the likelihood of the failure for SRVs to open, subsequently reduces the occurrence of this medium LOCA.	#4 - No significant safety benefit	The Medium LOCA frequency is 4.8E-05. The MSIV closure freq is 5.51E-2 per year. SRV common cause failure to open freq is 1.12E-6. Total contribution to LOCA is therefore 6.17E-8 or 0.1%, which is insignificant.		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
107	Install motor-driven feedwater pump.	1 12	SAMA would increase the availability of injection subsequent to MSIV closure.	Retain	PBAPS has 3 turbine driven feedwater pumps. This SAMA would increase high pressure make-up capability for scenarios where re-opening of the MSIVs is either not desirable or not proceduralized.		16
108	Enhance procedure to instruct operators to trip unneeded RHR/CS pumps on loss of room ventilation.	12	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.	Retain			17
109	Increase available net positive suction head (NPSH) for injection pumps.	1	SAMA increases the probability that these pumps will be available to inject coolant into the vessel by increasing the available NPSH for the injection pumps.	#3 - Already installed.	NPSH available can be increased by 1) increasing the levels in the CST and torus. 2) Containment pressure venting 3) Quality of water 4) Cue 5) Temperature. HPSW can be used to inject into the torus. CST can make-up to the torus and vice versa.	T-231-2 T-230-2 T-233-2	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
110	Increase the safety relief valve (SRV) reseat reliability.	1	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after standby liquid control (SLC) injection.	Retain			18
111	Reduce DC dependency between high-pressure injection system and ADS.	1	SAMA would ensure containment depressurization and high-pressure injection upon a DC failure.	#3 - Already installed.	ADS requires either 125 V DC Bus 20D21 or 125 V DC Bus 20D24. RCIC requires 125 V DC Bus 20D21 and bus 20D23. HPCI requires 125 V DC Bus 20D22 and 20D24. Loss of a single DC Bus can not disable ADS AND high pressure make-up systems.		N/A
112	Modify Reactor Water Cleanup (RWCU) for use as a decay heat removal system and proceduralize use.	1	SAMA would provide an additional source of decay heat removal.	Retain	Proceduralizing the use of RWCU as a decay heat removal system could be cost-effective. However, RWCU heat removal capacity may be low.		19
113	Use control rod drive (CRD) for alternate boron injection.	1 14	SAMA provides an additional system to address ATWS with SLC failure or unavailability.	#3 - Already installed.	The CRD can be aligned to take suction from the SBLC tank to allow for alternate boron injection into the RPV.	Procedure T-210-2	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
114	Increase seismic ruggedness of plant components.	11 13 14	SAMA would increase the availability of necessary plant equipment during and after seismic events.	#3 - Already installed.	Many components were identified in the IPEEE and SQUG programs whose seismic ruggedness could be improved. These items have been addressed in response to those efforts and satisfy the intent of this SAMA.	Evaluation of Peach Bottom Accident Management Insights	N/A
115	Allow cross connection of uninterruptable compressed air supply to opposite unit.	12 13	SAMA would increase the ability to vent containment using the hardened vent.	#3 - Already installed.	Vent depends on Instrument Air that can be cross-tied to other unit.	PBAPS PRA	N/A
116	Enhance RPV depressurization capability	14 15	SAMA would decrease the likelihood of core damage in loss of high pressure coolant injection scenarios	#3 - Already installed.	At PBAPS all SRVs have two redundant 125 VDC power supplies. The ADS nitrogen supply valves are powered from emergency buses. The ADS nitrogen supply is backed by bottles and an outside connection for long term nitrogen supply.	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
117	Enhance RPV depressurization procedures	14 15	SAMA would decrease the likelihood of core damage in loss of high pressure coolant injection scenarios	#3 - Already installed.	Both the EOP TRIP and SAMP procedures recognize the benefit of depressurization and 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 LOOP SE-11 procedure recognizes the need to provide emergency power to the ADS valves.	Evaluation of Peach Bottom Accident Management Insights	N/A
118	Bypass MSIV isolation in Turbine Trip ATWS scenarios	14	SAMA will afford operators more time to perform actions. 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	#3 - Already installed.	BWROG EPC Issue 98-07 addresses this issue. The bypass of the MSIV isolation was moved upward in the flowchart. PBAPS implementation has followed the BWROG recommendation in placement of this step	Evaluation of Peach Bottom Accident Management Insights	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
119	Enhance operator actions during ATWS	14	SAMA will reduce human error probabilities during ATWS	#3 - Already installed.	Operator actions during ATWS scenarios are clearly directed in the EOP TRIP procedures and receive attention in training.	Evaluation of Peach Bottom Accident Management Insights	N/A
120	Refill CST	14 16	SAMA 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.	#3 - Already installed.	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.	Evaluation of Peach Bottom Accident Management Insights	N/A
121	Maintain ECCS suction on CST	14 16	SAMA would maintain suction on the CST as long as possible to avoid pump failure as a result of high suppression pool temperature	#3 - Already installed.	Swap to/from CST source is procedurally directed.	Evaluation of Peach Bottom Accident Management Insights	N/A
122	Early detection and mitigation of ISLOCA	14 16	SAMA would limit the effects of ISLOCA accidents by early detection and isolation	#4 - No significant safety benefit	Related to mitigation of an ISLOCA. Per IN-92-36, and its additional supplement, ISLOCA contributes little risk for BWRs, because of the lower primary system pressures.	IN-92-36, and its additional supplement	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
123	CRD Injection	14 16	SAMA would supply an additional method of level restoration by using a non-safety system.	#3 - Already installed.	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.	Evaluation of Peach Bottom Accident Management Insights	N/A
124	Condensate Pumps for Injection	14 16	SAMA to provide an additional option for coolant injection when other systems are unavailable or inadequate	#3 - Already installed.	The use of condensate is covered in existing EOPs and in training.	Evaluation of Peach Bottom Accident Management Insights	N/A
125	Align EDG to CRD	14 16	SAMA to provide power to an additional injection source during loss of power events	#3 - Already installed.	CRD pumps at PBAPS are normally fed from diesel-backed emergency 4 kV buses.	Evaluation of Peach Bottom Accident Management Insights	N/A
126	Guard against SLC dilution	14 16	SAMA to control vessel injection to prevent boron loss or dilution following SLC injection.	#3 - Already installed.	SLC initiation and existing procedures guard against dilution (RWCU isolation and overfill prevention).	Evaluation of Peach Bottom Accident Management Insights	N/A
127	Re-open MSIVs	14 16	SAMA to regain the main condenser as a heat sink by re-opening the MSIVs.	#3 - Already installed. (also see 118)	Existing EOPs direct this including bypass of low level interlocks as necessary.	Evaluation of Peach Bottom Accident Management Insights	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
128	Bypass RCIC Turbine Exhaust Pressure Trip	14 16	SAMA would allow RCIC to operate longer.	#4 - No significant safety benefit	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.	Evaluation of Peach Bottom Accident Management Insights	N/A
129	Bypass Diesel Generator Trips	14 16	SAMA would allow D/Gs to operate for longer.	#3 - Already installed.	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.	Evaluation of Peach Bottom Accident Management Insights	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
130	Shield electrical equipment from potential water spray	14	SAMA would decrease risk associated with seismically induced internal flooding	#3 - Already installed.	A modification was identified for installation of a drip shield to protect inverter 20D37 from inadvertent spray. No additional modifications to EPGs/SAGs or other plant procedures (or equipment) are judged necessary.	Evaluation of Peach Bottom Accident Management Insights	N/A
131	Replace mercury switches on fire protection systems	14	SAMA would decrease probability of spurious fire suppression system actuation given a seismic event+D114	#3 - Already installed.	The U2 and U3 Reactor Building Water Curtain system manual pull stations have been replaced by manually operated switches. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A
132	Provide additional restraints for CO_2 tanks	14	SAMA would increase availability of fire protection given a seismic event.	#3 - Already installed.	Modifications to provide additional restraints for CO_2 tanks 00S101, 20S101, 30S101, and 20S112 have been performed. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A
133	Enhance control of transient combustibles	14	SAMA would minimize risk associated with important fire areas.	#3 - Already installed.	Procedures to control the transportation of combustible material are in place at Peach Bottom. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A
134	Enhance fire brigade awareness	14	SAMA would minimize risk associated with important fire areas.	#3 - Already installed.	Fire brigade awareness is in place at Peach Bottom. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
135	Upgrade fire compartment barriers	14	SAMA would minimize risk associated with important fire areas.	#3 - Already installed.	PBAPS fire compartment barriers have been improved to reduce fire propagation. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A
136	Enhance procedures to allow specific operator actions	14	SAMA would minimize risk associated with important fire areas.	#3 - Already installed.	Peach Bottom procedures have been enhanced. Based on IPEEE insights.	Evaluation of Peach Bottom Accident Management Insights	N/A
137	Develop procedures for transportation and nearby facility accidents	14	SAMA would minimize risk associated with transportation and nearby facility accidents.	#4 - No significant safety benefit	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	Evaluation of Peach Bottom Accident Management Insights	N/A
138	Enhance procedures to mitigate Large LOCA	14	SAMA would minimize risk associated with Large LOCA	#3 - Already implemented.	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.	Evaluation of Peach Bottom Accident Management Insights	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
139	Modify containment flooding procedure to restrict flooding to below TAF	14	SAMA would avoid forcing containment venting	#3 - Already implemented.	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	Evaluation of Peach Bottom Accident Management Insights	N/A
140	Enhance containment venting procedures with respect to timing, path selection and technique.	14	SAMA would improve likelihood of successful venting strategies.	#3 - Already implemented.	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	Evaluation of Peach Bottom Accident Management Insights	N/A
141	1.a. Severe Accident EPGs/AMGs	17	SAMA would lead to improved arrest of core melt progress and prevention of containment failure	#3 - Already implemented.	Latest revision of SAGs implemented. Also, additional procedural items addressed in other specific SAMAs (e.g., 20, 42).	PBAPS EOPs/SAMGs	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
142	1.b. Computer Aided Instrumentation	17	SAMA will improve prevention of core melt sequences by making operator actions more reliable.	#5 - ABWR Design Issue; not practical.	This is a SAMA which was considered for ABWR design. It is not practical to backfit this modification into a plant which is already built and operating. Also, see Table 6 and Section A.4.1.2 of Reference 17.	GE ABWR SAMDAs	N/A
143	1.c/d. Improved Maintenance Procedures/Manuals	17	SAMA will improve prevention of core melt sequences by increasing reliability of important equipment	#3 - Already implemented.	See Table 6 and Section A.4.1.3 of ABWR SAMDAs. Maintenance rule practices have helped evolve the performance of maintenance activities and have improved procedures and training.	GE ABWR SAMDAs	N/A
144	Not used	N/A	N/A	N/A	#N/A	N/A	N/A
145	1.e. Improved Accident Management Instrumentation	17	SAMA will improve prevention of core melt sequences by making operator actions more reliable.	#2 - Similar item is addressed under other proposed SAMAs.	Part of 142		N/A
146	1.f. Remote Shutdown Station	17	This SAMA would allow alternate system control in the event that the control room becomes uninhabitable.	#3 - Already implemented.	PBAPS already has a remote shutdown station.	PBAPS PRA	N/A
147	1.g. Security System	17	This SAMA would reduce the potential for sabotage.	#3 - Already implemented.	Electronic safety measures and trained security personnel provide surveillance for the PBAPS site.		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
148	1.h. Simulator Training for Severe Accident	17	SAMA would lead to improved arrest of core melt progress and prevention of containment failure	#2 - Similar item is addressed under other proposed SAMAs.	Training provided as part of 141		N/A
149	2.a. Passive High Pressure System	17	SAMA will improve prevention of core melt sequences by providing additional high pressure capability to remove decay heat through an isolation condenser type system	Retain	See Table 6 and Section A.4.2.1 of ABWR SAMDAs.		20
150	2.b. Improved Depressurization	17	SAMA will improve depressurization system to allow more reliable access to low pressure systems.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed in SAMAs 106, 116 and 117		N/A
151	2.c. Suppression Pool Jockey Pump	17	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.	Retain	Section A.4.2.3 - Similar to firewater injection and spray capability (#46), but it would have the advantage that long term containment inventory concerns would not occur.		21
152	2.d. Improved High Pressure Systems	17	SAMA will improve prevention of core melt sequences by improving reliability of high pressure capability to remove decay heat.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed in SAMA 107		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
153	2.e. Additional Active High Pressure System	17	SAMA will improve reliability of high pressure decay heat removal by adding an additional system.	Retain			22
154	2.f. Improved Low Pressure System (Firepump)	17	SAMA would provide fire protection system pump(s) for use in low pressure scenarios.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed in SAMA 46		N/A
155	2.g. Dedicated Suppression Pool Cooling	17	SAMA would decrease the probability of loss of containment heat removal.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 29		N/A
156	2.h. Safety Related Condensate Storage Tank	17	SAMA will improve availability of CST following a Seismic event	Retain	See Table 6 and Section A.4.2.4 of ABWR SAMDAs.		23
157	2.i. 16 hour Station Blackout Injection	17	SAMA includes improved capability to cope with longer station blackout scenarios.	#2 - Similar item is addressed under other proposed SAMAs.	Part of 197		N/A
158	Not used	N/A	N/A	N/A	N/A	N/A	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
159	3.a. Larger Volume Containment	17	SAMA increases time before containment failure and increases time for recovery	#2 - Similar item is addressed under other proposed SAMAs.	SAMA 52 addresses this issue.		N/A
160	3.b. Increased Containment Pressure Capability (sufficient pressure to withstand severe accidents)	17	SAMA minimizes likelihood of large releases	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 49		N/A
161	3.c. Improved Vacuum Breakers (redundant valves in each line)	17	SAMA reduces the probability of a stuck open vacuum breaker.	Retain	See Table 6 and Section A.4.3.3 of ABWR SAMDAs.		24
162	3.d. Increased Temperature Margin for Seals	17	This SAMA would reduce the potential for containment failure under adverse conditions.	#2 - Similar item is addressed under other proposed SAMAs.	Part of 160 (increased containment pressure capability)		N/A
163	3.e. Improved Leak Detection	17	The intent of this SAMA is to increase piping surveillance in order to identify leaks prior to the onset of complete failure. Improved leak detection would potentially reduce the LOCA frequency.	#1 - N/A	Containment inerting obviates the need for leak detection.	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
164	3.f. Suppression Pool Scrubbing	17	This SAMA would reduce the consequences of venting the containment by directing the ventpath through the water contained in the suppression pool.	#3 - Already implemented.	The PBAPS Torus Vent in located in the Wetwell airspace.	PBAPS PRA	N/A
165	3.g. Improved Bottom Penetration Design	17	SAMA reduces failure likelihood of RPV bottom head penetrations	#5 - ABWR Design Issue; not practical.	This is a SAMA which was considered for ABWR design. It is not practical to backfit this modification into a plant which is already built and operating.	ABWR SAMDAs	N/A
166	4.a. Larger Volume Suppression Pool (double effective liquid volume)	17	SAMA would increase the size of the suppression pool so that heatup rate is collapsed, allowing more time for recovery of a heat removal system	#5 - ABWR Design Issue; not practical.	This is a SAMA which was considered for ABWR design. It is not practical to backfit this modification into a plant which is already built and operating.	ABWR SAMDAs	N/A
167	4.b. CUW Decay Heat Removal	17	This SAMA provides a means for Alternate Decay Heat Removal.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 112. The CUW system in ABWR is equivalent to the RWCU system.		N/A
168	4.c. High Flow Suppression Pool Cooling	17	SAMA would improve suppression pool cooling.	#3 - Already implemented.	The Suppression Pool Cooling system is already sized to accommodate flow to remove all decay heat and operate under ATWS conditions.	PBAPS PRA	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
169	4.d. Passive Overpressure Relief	17	This SAMA will prevent catastrophic failure of the containment. Controlled relief through a selected vent path has a greater potential for reducing the release of radioactive material than through a random break.	#3 - Already implemented.	The Torus Vent is equipped with a rupture disk.	PBAPS PRA	N/A
170	5.a/d. Unfiltered Vent	17	SAMA would provide an alternate decay heat removal method with the released fission products not being scrubbed.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 32		N/A
171	5.b/c. Filtered Vent	17	SAMA would provide an alternate decay heat removal method with the released fission products being scrubbed.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 33 and 47		N/A
172	6.a. Post Accident Inerting System	17	SAMA would reduce likelihood of gas combustion inside containment	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 45		N/A
173	6.b. Hydrogen Control by Venting	17	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by venting the hydrogen gas prior to reaching detonable concentration.	#3 - Already implemented.	Addressed in EPGs/SAMGs	PBAPS EOPs/SAMGs	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
174	6.c. Pre-inerting	17	SAMA would reduce likelihood of gas combustion inside containment	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 45		N/A
175	6.d. Ignition Systems	17	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by burning the hydrogen gas prior to reaching detonable concentration.	#1 - N/A	Not applicable, since containment is inerted.	PBAPS PRA	N/A
176	6.e. Fire Suppression System Inerting	17	This SAMA will prevent catastrophic failure of the containment due to hydrogen detonation by inerting the containment with the fire suppression system.	#1 - N/A	Not applicable, since containment is inerted.	PBAPS PRA	N/A
177	7.a. Drywell Head Flooding	17	SAMA would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 39		N/A
178	7.b. Containment Spray Augmentation	17	SAMA would provide a redundant source of water to the containment to control containment pressure when used in conjunction with containment heat removal.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 30, 31		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
179	8.a. Additional Service Water Pump	17	SAMA might conceivably reduce common cause dependencies from SW system and thus reduce plant risk through system reliability improvement.	#2 - Similar item is addressed under other proposed SAMAs.	Although this SAMA is not directly addressed elsewhere, SAMAs 21 and 73 suggest using Fire Pumps as alternate service water sources.		N/A
180	8.b. Improved Operating Response	17	This SAMA would improve the likelihood of success of operator actions taken in response to an abnormal condition.	#3 - Already implemented.	Operator response has been a focus at PBAPS over the past decade. Training has been improved and procedures have been re-written in an ongoing effort to improve operator reliability.		N/A
181	8.c. Diverse Injection System	17	SAMA will improve prevention of core melt sequences by providing additional injection capabilities.	#2 - Similar item is addressed under other proposed SAMAs.	Part of 149, 153		N/A
182	8.d. Operation Experience Feedback	17	This SAMA would provide information on the effectiveness of maintenance practices and equipment reliability.	#3 - Already implemented.	Operational experienced is tracked and incorporated into future plant operating philosophy via programs such as the maintenance rule. Already incorporated at PBAPS.		N/A
183	8.e. Improved MSIV Design	17	This SAMA would decrease the likelihood of containment bypass scenarios.	Retain			25

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
184	8.e. Improved SRV Design	17	This SAMA would improve SRV reliability, thus increasing the likelihood that sequences could be mitigated using low pressure heat removal.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 106, 110		N/A
185	9.a. Steam Driven Turbine Generator	17	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.	Retain	See Table 6 and A.4.9.1 of ABWR SAMDAs		26
186	9.b. Alternate Pump Power Source	17	This SAMA would provide a small dedicated power source such as a dedicated diesel or gas turbine for the feedwater or condensate pumps, so that they do not rely on offsite power.	#2 - Similar item is addressed under other proposed SAMAs.	Firewater pump provides low pressure injection without offsite power (#46). Additional or passive high pressure systems addressed in other SAMAs, as is motor driven FW pump.		N/A
187	9.d. Additional Diesel Generator	17	SAMA would reduce the SBO frequency.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 54, 56		N/A
188	9.e. Increased Electrical Divisions	17	SAMA would provide increased reliability of AC power system to reduce core damage and release frequencies.	#5 - ABWR Design Issue; not practical.	This is a SAMA which was considered for ABWR design. It is not practical to backfit this modification into a plant which is already built and operating.	GE ABWR SAMDAs	N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
189	9.f. Improved Uninterruptable Power Supplies	17	SAMA would provide increased reliability of power supplies supporting front-line equipment, thus reducing core damage and release frequencies.	Retain			27
190	9.g. AC Bus Cross-Ties	17	SAMA would provide increased reliability of AC power system to reduce core damage and release frequencies.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 60, 64		N/A
191	9.h. Gas Turbine	17	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 70		N/A
192	9.i. Dedicated RHR (bunkered) Power Supply	17	This SAMA would improve the reliability of the RHR system by enhancing the AC power supply system.	Retain			28
193	10.a. Dedicated DC Power Supply	17	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).	Retain			29

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
194	10.b. Additional Batteries/Divisions	17	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).	#2 - Similar item is addressed under other proposed SAMAs.	Part of 193		N/A
195	10.c. Fuel Cells	17	SAMA would extend DC power availability in an SBO.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 58		N/A
196	10.d. DC Cross-ties	17	This SAMA would improve DC power reliability.	Retain	Only partially addressed by SAMA 61		30
197	10.e. Extended Station Blackout Provisions	17	SAMA would provide reduction in SBO sequence frequencies.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 57, 62, 63, 26, 195, 54, 67, 69		N/A
198	11.a. ATWS Sized Vent	17	This SAMA would be provide the ability to remove reactor heat from ATWS events.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 34		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
199	11.b. Improved ATWS Capability	17	This SAMA includes items which reduce the contribution of ATWS to core damage and release frequencies.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed by SAMAs 113, 118, 119		N/A
200	12.a. Increased Seismic Margins	17	This SAMA would reduce the risk of core damage and release during seismic events.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 76, 114		N/A
201	12.b. Integral Basemat	17	This SAMA would improve containment survivability under severe seismic activity.	#1 - N/A	Not applicable to PBAPS design	GE ABWR SAMDAs	N/A
202	13.a. Reactor Building Sprays	17	This SAMA provides the capability to use firewater sprays in the reactor building to mitigate release of fission products into the Rx Bldg following an accident.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 40		N/A
203	13.b. System Simplification	17	This SAMA is intended to address system simplification by the elimination of unnecessary interlocks, automatic initiation of manual actions or redundancy as a means to reduce overall plant risk.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed by SAMAs 12, 72, 78, 96, 106, 109, 111		N/A

Table G.4-1 (Cont'd)
Phase I SAMA

Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Screening Criteria	Disposition	Disposition Reference	Phase II SAMA ID number
204	13.c. Reduction in Reactor Building Flooding	17	This SAMA reduces the Reactor Building Flood Scenarios contribution to core damage and release.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMAs 97, 99		N/A
205	14.a. Flooded Rubble Bed	17	SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.	#2 - Similar item is addressed under other proposed SAMAs.	See SAMA 38		N/A
206	14.b. Reactor Cavity Flooder	17	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	#2 - Similar item is addressed under other proposed SAMAs.	Addressed in SAMAs 41 & 51		N/A
207	14.c. Basaltic Cements	17	SAMA minimizes carbon dioxide production during core concrete interaction.	#5 - ABWR Design Issue; not practical.	This is a SAMA which was considered for ABWR design. It is not practical to backfit this modification into a plant which is already built and operating.	ABWR SAMDAs	N/A

Notes to Table G.4-1:

1. NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," Volume 2, NRC, December 1997.

2. Letter from Mr. M. O. Medford (Tennessee Valley Authority) to NRC Document Control Desk, dated September 1, 1992, "Watts Bar Nuclear Plant Units 1 and 2 – Generic Letter (GL) – Individual Plant Examination (IPE) for Severe Accident Vulnerabilities – Response".

3. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Volume 1, Table 5.36 Listing of SAMDAs considered for the Comanche Peak Steam Electric Station, NRC, May 1996.

4. Letter from Mr. D. E. Nunn (Tennessee Valley Authority) to NRC Document Control Desk, dated October 7, 1994, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDA) – Response to Request for Additional Information (RAI)".

5. "Cost Estimate for Severe Accident Mitigation Design Alternatives, Limerick Generating Station for Philadelphia Electric Company," Bechtel Power Corporation, June 22, 1989.

6. NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants," Volume 1, Table 5.35, Listing of SAMDAs considered for the Limerick, NRC, May 1996.

7. Letter from Mr. W. J. Museler (Tennessee Valley Authority) to NRC Document Control Desk, dated October 7, 1994, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDA)."

8. NUREG-0498, "Final Environmental Statement related to the operation of Watts Bar Nuclear Plant, Units 1 and 2," Supplement No. 1, NRC, April 1995.

9. Letter from Mr. D. E. Nunn (Tennessee Valley Authority) to NRC Document Control Desk, dated June 30, 1994. "Watts Bar Nuclear Plant (WBN) Unit 1 and 2 – Severe Accident Mitigation Design Alternatives (SAMDAs) Evaluation from Updated Individual Plant Evaluation (IPE)."

10. Letter from N. J. Liparulo (Westinghouse Electric Corporation) to NRC Document Control Desk, dated December 15, 1992, “Submittal of Material Pertinent to the AP600 Design Certification Review.”

11. NUREG-1462, “Final Safety Evaluation Report Related to the Certification of the System 80+ Design,” NRC, August 1994.

12. Hatch Individual Plant Examination.

13. Hatch Individual Plant Examination of External Events.

14. PBAPS Report on Accident Management Insights (includes disposition of IPE/PRA Level 1 and 2 insights and IPEEE insights)

15. GL 88-20, Supplement 1, NUREG-1335, “Individual Plant Examination: Submittal Guidance,” August 29, 1989.

16. GL 88-20, Supplement 2, “Accident Management Strategies for Consideration in the IPE Process, ”April 4, 1990.

17. GE Nuclear Energy, "Technical Support Document for the ABWR," 25A5680, Rev. 1, November 1994.

<u>Screening Criteria for Table G.4-1</u>:

#1: Not applicable.

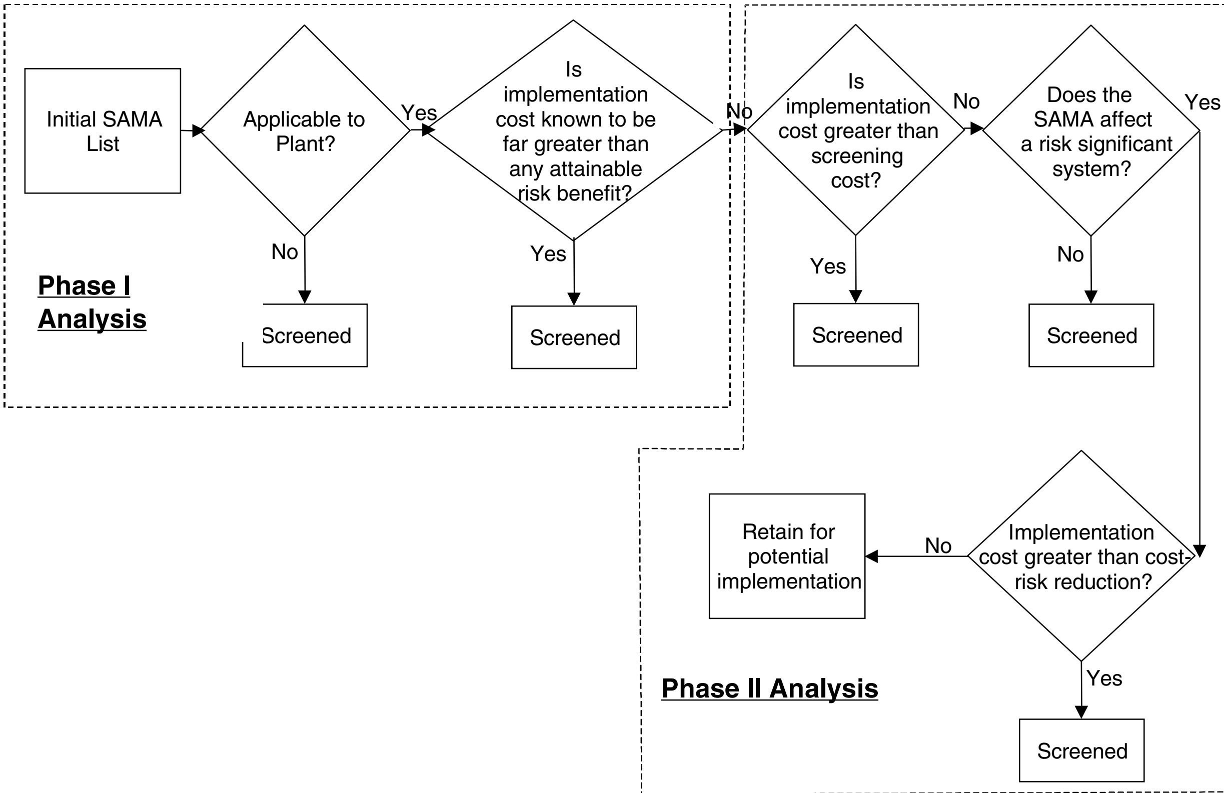
#2: Similar item is addressed under other proposed SAMA.

#3: Already installed.

#4: No significant safety benefit

#5: Cost would be more than risk benefit

Figure G.4-1
SAMA Screening Process



A majority of the SAMAs were removed from further consideration as they did not apply to the BWR-4/Mark I design used at PBAPS. An additional set of candidates was removed from consideration because all of those within the group were related to mitigation of an interfacing system Loss of Coolant Accident (ISLOCA). According to NRC Information Notice 92-36 and its supplement, ISLOCA contributes little risk for boiling water reactors because of the lower primary pressures. Review of the PBAPS PSA confirms that ISLOCA is a low contributor to risk (less than 0.1% of the internal CDF and less than 1.5% of internal LERF) and the risk benefit associated with improving ISLOCA mitigation in not significant. SAMA candidates related to Reactor Coolant Pump seal leakage were also removed from consideration. NUREG-1560 (Reference 5) indicates that although RCP seal leakage is important for PWRs, recirculation pump leakage does not significantly contribute to core damage frequency in BWRs.

The SAMA candidates that were found to be in place at PBAPS were screened from further consideration.

The SAMAs related to design changes prior to construction (primarily consisting of those candidates taken from the ABWR SAMAs) were removed as they were not practicable to an existing plant. For example, using basalfie cement (SAMA 207) would require dismantling of the reactor pedestal structure and replacement of the containment floor. This would result in exorbitant costs to implement. Any candidate known to have an implementation cost that far exceeds any possible risk benefit is screened from further analysis. Any SAMA candidates that were sufficiently similar to other SAMA candidates were treated in the same manner to those that they were related to; either combined or screened from further consideration. This screening left 30 unique SAMA candidates (Table G.4-2) that were potentially applicable to PBAPS and were of potential value in averting the risk of severe accidents. Section G.5 describes the process used to disposition the remaining SAMAs.

Section G.5 describes the results of the detailed cost benefit analysis.

TABLE G.4-2
PHASE II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
1	19	Enhance procedural guidance for use of cross-tied component cooling or service water pumps.	1 14	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.
2	21	Improved ability to cool the residual heat removal heat exchangers.	1	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.
3	29	Install an independent method of suppression pool cooling.	5 6	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 >$1M.	Screened ($)
4	33	Install a filtered containment vent to remove decay heat.	5 6	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	[$3M/Unit X 2] - Ref. G.8-17, Section A.5.5.1	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
5	34	Install a containment vent large enough to remove ATWS decay heat.	5 6	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	>$2M	[$300K/Unit x 2] - Ref. G.8-17, Section A.5.11.1, but installation of hard pipe vent at PB cost >$2 million (Ref. G.8-18)	Screened ($)
6	46	Use the fire protection system as a backup source for the containment spray system.	4	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.
7	48	Install a passive containment spray system.	10	SAMA would provide redundant containment spray method without high cost.	>$2M	Assumed to be similar in cost to passive HP system (SAMA 149)	Screened ($)
8	52	Construct a building to be connected to primary/secondary containment that is maintained at a vacuum.	11	SAMA would provide a method to depressurize containment and reduce fission product release.	>$2M	$'s per engineering judgment	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
9	54	Proceduralize alignment of spare diesel to shutdown board after loss of offsite power and failure of the diesel normally supplying it.	1 3 7	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 ($)
10	56	Provide an additional diesel generator.	1 3 7 11 14	SAMA would increase the reliability and availability of onsite emergency AC power sources.	>$2M	$'s per engineering judgment. Ref. G.8-17 lists cost at approximately $1.2M. However, this is significantly less than cost of installing new DGs after plant is built (Calvert Cliffs >$100M for 2 new DGs).	Screened ($)
11	57	Provide additional DC battery capacity.	1 3 7 11 12	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.
12	58	Use fuel cells instead of lead-acid batteries.	11	SAMA would extend DC power availability in an SBO.	>$2M	[$6M] - Ref. G.8-17, Section A.5.10.1	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
13	66	Develop procedures to repair or replace failed 4-kV breakers.	1	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.
14	70	Install gas turbine generator.	1 14	SAMA would improve onsite AC power reliability by providing a redundant and diverse emergency power system.	>$2M	$'s per engineering judgment	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
15	105	Proceduralize intermittent operation of HPCI.	1	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).
16	107	Install motor-driven feedwater pump.	1 12	SAMA would increase the availability of injection subsequent to MSIV closure.	>$2M	$'s per engineering judgment	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
17	108	Enhance procedure to instruct operators to trip unneeded RHR/CS pumps on loss of room ventilation.	12	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.
18	110	Increase the safety relief valve (SRV) reseat reliability.	1	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.
19	112	Modify Reactor Water Cleanup (RWCU) for use as a decay heat removal system and proceduralize use.	1	SAMA would provide an additional source of decay heat removal.	>$2 million for hardware upgrade	RWCU heat removal capacity is 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.

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
20	149	2.a. Passive High Pressure System	17	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	[$1.7M x 2] - Ref. G.8-17, Section A.5.2.1	Screened ($)
21	151	2.c. Suppression Pool Jockey Pump	17	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. G.8-17, 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.
22	153	2.e. Additional Active High Pressure System	17	SAMA will improve reliability of high pressure decay heat removal by adding an additional system.	>$2M	Assumed to be similar in cost to passive HP system (SAMA 149)	Screened ($)
23	156	2.h. Safety Related Condensate Storage Tank	17	SAMA will improve availability of CST following a Seismic event	>$2M	[>$1M x 2] - Ref. G.8-17, Section A.5.2.4	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
24	161	3.c. Improved Vacuum Breakers (redundant valves in each line)	17	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 >$1M/unit.	Screened ($)
25	183	8.e. Improved MSIV Design	17	This SAMA would decrease the likehood of containment bypass scenarios.	>$2M	Assume $200K/MSIV x 16 MSIVs (8 per unit)	Screened ($)
26	185	9.a. Steam Driven Turbine Generator	17	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	[$6M x 2} - Ref. G.8-17, Section A.5.9.1	Screened ($)
27	189	9.f. Improved Uninterruptable Power Supplies	17	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.

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
28	192	9.i. Dedicated RHR (bunkered) Power Supply	17	This SAMA would improve the reliability of the RHR system by enchancing the AC power supply system.	>$2M	[$1.2M x 2] - Ref. G.8-17, Section A.5.9.2	Screened ($)
29	193	10.a. Dedicated DC Power Supply	17	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	[$3M x 2] - Ref. G.8-17, Section A.5.10.1	Screened ($)

Table G.4-2 (Cont'd)
Phase II SAMA

Phase II SAMA ID number	Phase I SAMA ID number	SAMA title	Source Reference of SAMA [See Notes]	Result of potential enhancement	Estimated Cost	Comment	Phase II Disposition
30	196	10.d. DC Cross-ties	17	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.

G.5 PHASE II SAMA ANALYSIS

A preliminary cost estimate was prepared for each of the remaining candidates to focus on those that had the possibility of having a positive benefit and to eliminate those whose costs were beyond the possibility of any corresponding benefit. When the screening cutoff of $2,040,468 was applied, 18 candidates were eliminated that were more expensive than the maximum postulated benefit associated with the elimination of all risk associated with full power internal events. This left 12 candidates for further analysis. Those SAMAs which required a more detailed cost benefit analysis were evaluated using the combined methods described in Sections G.2 and G.3. Other SAMA candidates were screened from further analysis based on plant specific insights regarding the risk significance of the systems that would be affected by the proposed SAMAs. The SAMAs related to non-risk significant systems were screened from a detailed cost benefit analysis as any change in the reliability of these systems is known to have a negligible impact on the PSA evaluation.

For each of the remaining SAMA candidates not eliminated based on screening cost or PSA/application insights, a more detailed conceptual design was prepared along with a more detailed estimated cost. This information was then used to evaluate the candidates' effects on the plant safety model.

The final cost-risk based screening method used to determine the desirability of implementing the SAMA is defined by the following equation:

Net Value = (baseline cost-risk of plant operation – cost-risk of plant operation with SAMA implemented) – cost of implementation

If the net value of the SAMA is negative, the cost of implementation is larger than the benefit associated with the SAMA and the SAMA is not considered beneficial. The baseline cost-risk of plant operation was derived using the methodology presented in Section G.3. The cost-risk of plant operation with the SAMA implemented is determined in the same manner with the exception that the PSA results reflect the application of the SAMA to the plant (the baseline input is replaced by the results of a PSA sensitivity with the SAMA change in effect).

Subsections G.5.1 – G.5.5 describe the detailed cost-benefit analysis that was used to determine how the remaining candidates were ultimately treated. The results are presented on a site (2 units) basis.

G.5.1 PHASE II SAMA NUMBER 1, ENHANCE PROCEDURAL GUIDANCE FOR USE OF CROSS-TIED COMPONENT COOLING OR SERVICE WATER PUMPS

Description: In this sensitivity, it was assumed that the guidance would virtually eliminate initiating events related to loss of service water. For PBAPS, this was assumed to relate to the loss of service water initiating event, the loss of TBCCW initiating event, and the loss of RBCCW initiating event. This impact was chosen for the study because the importance of these systems from a mitigation perspective is already low and because the impact of improving their reliabilities would maximize the calculated benefit by virtually eliminating these systems as initiating events.

To implement this change, the following basic event values were changed as indicated in Table G.5.1-1 in the PBAPS Unit 2 model to simulate almost totally reliable service water systems from an initiating event perspective.

TABLE G.5.1-1
PHASE II SAMA NUMBER 1 MODEL CHANGES

System: Basic Events	Original Value	Revised Value
Service Water Pumps fail to run in 8760 hours:		
PPMAP04I2	0.231	0.00
PPMBP04I2	0.231	0.00
PPMCP04I2	0.231	0.00
TBCCW Pumps fail to run in 8760 hours:		
TPMA144I2	0.231	0.00
TPMB144I2	0.231	0.00
RBCCW Pumps fail to run in 8760 hours:		
BPMAP10I2	0.231	0.00
BPMBP10I2	0.231	0.00

PSA Model Results (Phase II SAMA Number 1)

The results from this case indicate about a 0.7% reduction in Unit 2 CDF (CDF_{new}=4.5E-6/yr) and a 0.2% reduction in LERF ($LERF_{new}$=6.2E-8/yr). The results of the cost-benefit analysis are shown in Table G.5.1-2.

TABLE G.5.1-2
PHASE II SAMA NUMBER 1 NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 1: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040468	$2,032,059	$8,409	$50,000	-$41,591

The negative net value of this SAMA candidate indicates that its implementation is not beneficial.

G.5.2 PHASE II SAMA NUMBER 11, PROVIDE ADDITIONAL DC BATTERY CAPACITY

Description: In this sensitivity, it was assumed that the battery life could be extended to 4 hours each to simulate additional battery capacity. The 4 hour battery life could be obtained by installing improved batteries. This enhancement would impact the loss of offsite power cases with HPCI and/or RCIC available (i.e., the Te1a, Te1b, Te2a, Te2b, Te3a, Te3b, Te5a, and Te5b event trees). With HPCI or RCIC available, but with no AC power to the corresponding battery charger that supports HPCI or RCIC operation, 2.5 hours is assumed to be available to recover offsite power based on two hours of battery life and one half hour of boildown time. The 2.5-hour assumption is changed to 5 hours in this SAMA case (4 hours of battery life and 1 hour for boildown). Correspondingly, with both HPCI and RCIC available, but no AC power to the corresponding battery chargers, 5 hours is assumed to be available to recover offsite power before both HPCI and RCIC are lost due to loss of DC (4 hours of battery life and 1 hour for boildown). The 5-hour assumption is changed to 10 hours in this SAMA case (8 hours of battery life and 2 hours for boildown. Containment heat removal is also assumed to be necessary).

Table G.5.2-1 summarizes the changes made in the PBAPS Unit 2 PSA model to simulate the effects of this SAMA.

TABLE G.5.2-1
PHASE II SAMA NUMBER 11 MODEL CHANGES

Basic Event: Description	Original Value	Revised Value
ROSP2U Fail to recover offsite power Changed from 2.5 hour value to 5 hour value	0.225	0.113
ROSP5 Fail to recover offsite power Changed from 5 hour value to 10 hour value	0.113	0.041
NOSP10U Fail to recover at 10 hours given not recovered at 2.5. Changed from 10/2.5 value to 10/5 value	0.182	0.363
NOSP105 Fail to recover at 10 hours given not recovered at 2.5. Changed from 10/5 value to 10/10 value	0.363	1.0

PSA Model Results (Phase II SAMA Number 11)

The PSA results for this case indicate about a 19% reduction in Unit 2 CDF (CDF_{new}= 3.7E-6/yr) and a 10% reduction in LERF ($LERF_{new}$=5.6E-8/yr). The results of the cost-benefit analysis for Phase II SAMA 11 are shown in Table G.5.2-2.

TABLE G.5.2-2
PHASE II SAMA NUMBER 11 NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 11: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040,468	$1,775,371	$265,097	$1,600,000	-$1,334,903

The negative net value of this SAMA candidate (installation of new batteries) indicates that its implementation is not beneficial.

G.5.3 PHASE II SAMA NUMBER 13, DEVELOP PROCEDURES TO REPAIR OR REPLACE FAILED 4-KV BREAKERS

Description: In this model run, it was assumed that the improved procedures to repair or replace failed 4 kV breakers would result in collapsed 4 kV breaker "fail to close rates". However, since these failures only manifest themselves in the model for implementation of the PBAPS SE-11 procedure for cross-tying buses, an additional change was also made to the 4 kV bus failure rates to further simulate the improved performance that could be obtained from this SAMA.

To implement this change, basic event values were changed as indicated in Table G.5.3-1 in the PBAPS Unit 2 model to simulate alternate 4-kV breaker capability.

TABLE G.5.3-1
PHASE II SAMA NUMBER 13 MODEL CHANGES

System: Basic Events	Original Value	Revised Value
4 kV Circuit Breakers fail to close:		
ECB1505N2	5.0×10^{-4}	0.00
ECB1505N3	5.0×10^{-4}	0.00
ECB1605N2	5.0×10^{-4}	0.00
ECB1605N3	5.0×10^{-4}	0.00
ECB1705N2	5.0×10^{-4}	0.00
ECB1705N3	5.0×10^{-4}	0.00
ECB1806N2	5.0×10^{-4}	0.00
ECB1806N3	5.0×10^{-4}	0.00
4 kV Buses fail:		
EBSA15XW2	2.4×10^{-6}	2.4×10^{-7}
EBSA15XW3	2.4×10^{-6}	2.4×10^{-7}
EBSA16XW2	2.4×10^{-6}	2.4×10^{-7}
EBSA16XW3	2.4×10^{-6}	2.4×10^{-7}
EBSA17XW2	2.4×10^{-6}	2.4×10^{-7}
EBSA17XW3	2.4×10^{-6}	2.4×10^{-7}
EBSA18XW2	2.4×10^{-6}	2.4×10^{-7}
EBSA18XW3	2.4×10^{-6}	2.4×10^{-7}

PSA Model Results (Phase II SAMA Number 13)

The results from this case indicate about a 0.1% reduction in CDF ($CDF_{new}=4.5\times10^{-6}$/yr) and a 0.1% reduction in LERF ($LERF_{new}=6.2\times10^{-6}$/yr). The results of the cost-benefit analysis are shown in Table G.5.3-2.

TABLE G.5.3-2
PHASE II SAMA NUMBER 13 NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 13: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040,468	$2,040,080	$388	$50,000	-$49,612

The negative net value of this SAMA candidate indicates that its implementation is not beneficial.

G.5.4 PHASE II SAMA NUMBER 18, INCREASE THE SAFETY RELIEF VALVE RE-SEAT RELIABILITY

Description: In this model run, it was assumed that the improved reliability of the SRVs would result in collapsed "fail to reseat" probabilities for the SRVs. This issue is included to address the risk associated with dilution of boron caused by the failure of the SRVs to re-seat after standby liquid control (SLC) injection. However, the improved reliability would impact non-ATWS cases as well in collapsed consequential stuck open relief valve scenarios, and in stuck open relief valve initiating events.

To implement this change, basic event values were changed as indicated in Table G.5.4-1 in the PBAPS Unit 2 model to simulate improved SRV re-seat reliability.

TABLE G.5.4-1
PHASE II SAMA NUMBER 18 MODEL CHANGES

System: Basic Events	Original Value	Revised Value
SRV(s) fail to re-seat (Included in SAMA Case 18a and 18b):		
P	7.99×10^{-2}	7.99×10^{-3}
P1	1.33×10^{-2}	1.33×10^{-3}
P2	2.66×10^{-2}	2.66×10^{-3}
P3	1.97x10-3	1.97×10^{-4}
P12	1.97×10^{-3}	1.97×10^{-4}
P22	1.97×10^{-6}	1.97×10^{-7}
P32	1.97×10^{-6}	1.97×10^{-7}
SORV Initiating Event (Included in SAMA Case 18b only):		
IETI	5.75×10^{-2}	5.75×10^{-3}

PSA Model Results (Phase II SAMA Number 18a)

The results from this case indicate about a 4% reduction in CDF ($CDF_{new}=4.4\times10^{-6}$/yr) and a 2% reduction in LERF ($LERF_{new}=6.0\times10^{-6}$/yr). The results of the cost-benefit analysis are shown in Table G.5.4-2.

TABLE G.5.4-2
PHASE II SAMA NUMBER 18A NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 18a: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040,468	$1,946,683	$93,785	$2,000,000	-$1,906,215

The negative net value of this SAMA candidate indicates that its implementation is not beneficial.

PSA Model Results (Phase II SAMA 18b)

The results from this case indicate about a 6% reduction in CDF (CDF_{new}=4.3×10^{-6}/yr) and a 2% reduction in LERF ($LERF_{new}$=6.0×10^{-8}/yr). The results of the cost-benefit analysis are shown in Table G.5.4-3.

TABLE G.5.4-3
PHASE II SAMA NUMBER 18B NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 18b: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040,468	$1,866,230	$174,238	$2,000,000	-$1,825,762

The negative net value of this SAMA candidate indicates that even if the improved SRV re-seat reliability also leads to a reduction in stuck open relief valve initiating events, its implementation is still not beneficial.

G.5.5 PHASE II SAMA NUMBER 21, INSTALL SUPPRESSION POOL JOCKEY PUMP FOR ALTERNATE INJECTION TO THE RPV

Description: In this model run, it was assumed that the installation of a suppression pool jockey pump would provide an independent means of providing long term injection to the RPV. Currently, the PBAPS model includes a simple representation of the fire pump to perform a similar function. Minimal credit is taken for success of the fire pump since it requires installation of separate cross-tie components. To simulate the potential impact of the dedicated jockey pump to perform this role, it was determined that the failure probability for the fire pump could be adjusted.

To implement this change, the following basic event value was changed as indicated in Table G.5.5-1 in the PBAPS Unit 2 model to simulate the incorporation of a dedicated independent system to provide injection from the suppression pool that could potentially be provided by the addition of a suppression pool jockey pump. The revised value of 0.01 is considered somewhat optimistic for the combined failure rate (including all dependencies and human error contribution) for this system. This optimistic value would lead to the maximum potential benefit from this SAMA.

TABLE G.5.5-1
PHASE II SAMA NUMBER 21 MODEL CHANGES

System: Basic Events	Original Value	Revised Value
Suppression Pool Jockey Pump fails: FIREPUMP	0.80	0.01

PSA Model Results (Phase II SAMA Number 21)

The results from this case indicate about an 8% reduction in CDF (CDF_{new}=4.2×10^{-6}/yr) and no reduction in LERF. While the PBAPS PSA results show no decrease in LERF, the translation of the PBAPS PSA model's Level 2 endstates into the collapsed APBs conservatively grouped "late" releases into the "early" bins due to the definition of the collapsed APBs. This is conservative and results in a more dramatic decrease in cost-risk than would be expected from the installation of the jockey pump considering the PBAPS PSA Level 2 model. The results of the cost-benefit analysis are shown in Table G.5.5-2.

TABLE G.5.5-2
PHASE II SAMA NUMBER 21 NET VALUE

Base Case: Cost-Risk for the PBAPS Site	SAMA 21: Cost-Risk for the PBAPS Site	Averted Cost-Risk	Cost of Implementation	Net Value
$2,040,468	$1,689,512	$350,956	$480,000	-$129,044

The negative net value of this SAMA candidate indicates that its implementation is not beneficial.

G.6 PHASE II SAMA ANALYSIS SUMMARY

The SAMA candidates not eliminated from consideration by the baseline screening process or other PSA insights required the performance of a detailed analysis of the averted cost-risk and SAMA implementation costs. SAMA candidates are judged to be justified modifications if the averted cost-risk resulting from the modification is greater that the cost of implementing the SAMA. Table G.6-1 summarizes the results of the detailed analyses that were performed for the SAMA candidates. None of the SAMAs analyzed were found to be cost-beneficial as defined by the methodology used in this study.

TABLE G.6-1
SUMMARY OF THE DETAILED SAMA ANALYSES

Phase II SAMA ID	Averted Cost-Risk	Cost of Site Implementation	Net Value
1	$8,409	$50,000	-$41,591
11	$265,097	$1,600,000	-$1,334,903
13	$388	$50,000	-$49,612
18(a)	$93,785	$2,000,000	-$1,906,215
18(b)	$174,238	$2,000,000	-$1,825,762
21	$350,956	$480,000	-$129,044

G.7 CONCLUSIONS

The results of this study indicate that none of the SAMA candidates would yield a significant reduction in public risk relative to the cost required to implement the SAMA. No plant changes or modifications have been identified for implementation or further review at PBAPS.

G.8 REFERENCES

Ref. G.8-1 NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U. S. Nuclear Regulatory Commission, Washington, D.C., June 1989.

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