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Attachment E

Severe Accident Mitigation Alternatives Analysis

Attachment E contains the following sections.

E.1 – Evaluation of PSA Model

E.2 – Evaluation of SAMA Candidates

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ATTACHMENT E.1

EVALUATION OF PSA MODEL

E.1 EVALUATION OF PROBABILISTIC SAFETY ANALYSIS MODEL

The severe accident risk was estimated using the Probabilistic Safety Analysis (PSA) model and a Level 3 model developed using the MACCS2 code. The CAFTA code was used to develop the Pilgrim Nuclear Power Station (PNPS) PSA Level 1 and Level 2 models. This section provides the description of PNPS PSA Levels 1, 2, and 3 analyses, Core Damage Frequency (CDF) uncertainty, Individual Plant Examination of External Events (IPEEE) analyses, and PSA model peer review.

E.1.1 PSA Model – Level 1 Analysis

The PSA model (Level 1 and Level 2) used for the SAMA analysis was the most recent internal events risk model for PNPS (Revision 1, April 2003) [Reference E.1-1]. The PNPS PSA model and documentation has been updated to reflect the current plant operating configuration and design changes as of September 2001. The current PSA model reflects the accumulation of additional plant operating history and component failure and unavailability data as of December 2001. The PSA model also resolves all findings and observations during the industry peer review of the model, conducted in March 2000 [Reference E.1-1]. The PNPS model adopts the small event tree/ large fault tree approach and uses the CAFTA code for quantifying CDF. The Level 1 and Level 2 PNPS PSA analyses were originally developed and submitted to the NRC in September 1992 as the Pilgrim Nuclear Power Station Individual Plant Examination (IPE) Submittal [Reference E.1-2].

The PSA model has been updated since the IPE due to the following.

- In 1995, the original IPE model was changed in response to the NRC Request for Additional Information (RAI) received in April 1995 [Reference E.1-3]. Overall CDF was reduced from 5.85E-5/yr to 2.84E-5/yr. The reduction in CDF was due to removal of HPCI room cooling dependency, revised ADS success criteria, and improved HPCI/RCIC performance.

- Equipment performance - As data collection progresses, estimated failure rates and system unavailability data change.

- Plant configuration changes - Plant configuration changes are incorporated into the PSA model.

- Modeling changes - The PSA model is refined to incorporate the latest state of knowledge and recommendations from internal and industry peer reviews.

The PSA model contains the major initiators leading to core damage with baseline CDFs listed in Table E.1-2 [Reference E.1-1].

The current PNPS PSA model was reviewed to identify those potential risk contributors that made a significant contribution to CDF. CDF-based Risk Reduction Worth (RRW) rankings were reviewed down to 1.005. Events below this point would influence the CDF by less than 0.5% and

are judged to be highly unlikely contributors for the identification of cost-beneficial enhancements. These basic events, including component failures, operator actions, and initiating events, were reviewed to determine if additional SAMA actions may need to be considered.

Table E.1-3 provides a correlation between the Level 1 RRW risk significant events (component failures, operator actions, and initiating events) down to 1.005 identified from the PNPS PSA model and the SAMAs evaluated in Section E.2.

The uncertainty associated with CDF was estimated using Monte Carlo techniques implemented in CAFTA for the base case mode. The results are shown in Table E.1-1.

Table E.1-1
Core Damage Frequency Uncertainty

Confidence	CDF (\RY)
Mean value	6.68E-6
5th percentile	4.30E-6
50th percentile	5.93E-6
95th percentile	1.08E-5

The values in Table E.1-1 reflect the uncertainties associated with the data distributions used in the analysis. The ratio of the 95th percentile to the mean is about 1.62. This uncertainty factor is included in the factor of 6 used to determine the upper bound estimated benefit described in Appendix E, Section 4.21.5.4.

Table E.1-2
PNPS PSA Model CDF Results by Major Initiators

IE Type	IE Description	CDF (/RY)	Percentage of CDF
TDC	Loss of DC Power Buses	3.06E-06	47.77%
LOOP	Loss of Offsite Power	1.29E-06	20.12%
TAC	Loss of AC Power Buses	8.83E-07	13.78%
LSSW	Loss of Salt Service Water	3.91E-07	6.10%
TRAN	Transients	3.60E-07	5.62%
LOCA	Loss of Coolant Accident	1.75E-07	2.73%
SBO	Station Blackout	1.46E-07	2.28%
ATWS	Anticipated Transient Without Scram	5.30E-08	0.83%
ISLOCA	Interfacing System LOCA	3.64E-08	0.57%
FLOOD	Internal Flooding	1.28E-08	0.20%
	Total	**6.41E-06**	**100.00%**

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
IE-T1	6.70E-02	1.337	Loss of offsite power (LOOP)	This term represents the LOOP initiating event. Industry efforts over the last twenty years have led to a significant reduction in plant scrams from all causes. Improvements related to enhancing offsite power availability or reliability and coping with SBO events were already implemented and evaluated during Phase I SAMA screening. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
IE-TDCB	2.63E-03	1.319	Transient caused by loss of 125VDC bus B	This term represents an initiating event caused by a complete loss of 125VDC buses D-17, D5, and D37 and random failures of battery D-2. Phase I SAMAs to improve battery charging capability and replace existing batteries with more reliable ones have already been installed. Phase II SAMAs 025, 026, 027, 031, 032, 033, 034, and 035 for enhancing DC system availability and reliability were evaluated.
IE-TDCA	2.63E-03	1.304	Transient caused by loss of 125VDC bus A	This term represents an initiating event caused by a complete loss of 125VDC buses D-16, D4, and D36, and random failures of battery D-1. Phase I SAMAs to improve battery charging capability and replace existing batteries with more reliable ones have already been installed. Phase II SAMAs 025, 026, 027, 031, 032, 033, 034, and 035 for enhancing DC system availability and reliability were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
FXT-XHE-FO-V4T2	2.31E-02	1.121	Operator fails to align fire water crosstie for reactor pressure vessel (RPV) injection via LPCI (transient)	This term represents operator failure to align fire water via the LPCI injection path for alternate RPV vessel injection. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. Phase II SAMAs 057 and 059, which recommend proceduralizing use of the diesel fire pump hydroturbine following EDG A failure, and providing a redundant path from fire water pump discharge to LPCI loops A and B cross-tie, were evaluated.
AC2-PHN-PE-23kV	5.00E-01	1.079	Loss of shutdown transformer 23kV feed	This term represents loss of the shutdown transformer 23kV feed to 4.16kV bus A8. Improvements related to enhancing offsite power availability or reliability and coping with SBO events were already implemented and evaluated during Phase I SAMA screening. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
IE-TSSW	6.85E-05	1.065	Loss of salt service water (SSW) system	This term represents an initiating event caused by a complete loss of the service water system. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
IE-TAC6	2.63E-03	1.059	Transient caused by loss of 4160VAC bus A6	This term represents loss of 4.16kV bus A6. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedures to repair or replace failed 4.16kV breakers have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
CIV-XHE-FO-DTV	3.01E-03	1.057	Operator fails to vent containment using direct torus vent (DTV)	This term represents operator failure to recognize the need to vent the torus for pressure reduction during loss of containment heat removal accident sequences. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. Phase II SAMA 053 to control containment venting within a narrow pressure band to prevent rapid containment depressurization during venting was evaluated.
IE-TAC5	2.63E-03	1.052	Transient caused by loss of 4160VAC bus A5	This term represents an initiating event caused by loss of 4.16kV bus A5. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedures to repair or replace failed 4.16kV breakers have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
RHR-MAI-MA-HTXAP	4.08E-04	1.051	RHR heat exchanger E-207A unavailable due to maintenance	This term represents RHR heat exchanger E-207A unavailable due to maintenance, leading to loop A RHR suppression pool cooling and drywell spray modes being unavailable for containment pressure reduction. Phase I SAMAs have already been implemented to use firewater for drywell spray and to use venting via DTV path to reduce containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated.
RBC-MAI-MA-LOOPA	3.71E-04	1.046	RBCCW loop A out for maintenance	This term represents RBCCW loop A unavailable due to maintenance. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
FXT-XHE-FO-DWS	2.21E-02	1.046	Operator fails to align fire water cross-tie for drywell spray	This term represents operator failure to align fire water via the LPCI injection path for alternate drywell spray to remove containment heat. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. Phase II SAMAs 057 and 059, which recommend proceduralizing use of the diesel fire pump hydroturbine following EDG A failure, and providing a redundant path from fire water pump discharge to LPCI loops A and B cross-tie, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
AC8-CBR-CO-204	9.50E-05	1.044	480V circuit breaker 52-204 fails to remain closed	This term represents random failure of 480V circuit breaker 52-204, leading to loss of power to 480V motor control center (MCC) B14 and its associated loads. A Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 030 and 058 to improve 480V bus availability were evaluated.
AC8-CBR-CO-103	9.50E-05	1.044	480V circuit breaker 52-103 fails to remain closed	This term represents random failure of 480V circuit breaker 52-103, leading to loss of power to 480V MCC B15 and its associated loads. A Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 030 and 058 to improve 480V bus availability were evaluated.
FXT-ENG-FR-P140	1.92E-02	1.043	Diesel fire pump P-140 fails to run	This term represents diesel fire pump P-140 failure to run. Phase II SAMA 045, to add a diverse injection system and provide an injection source other than fire water, was evaluated.
LCI-HTX-VF-E207A	3.24E-04	1.04	Loop B heat exchanger E-207A failure	This term represents random failure of RHR heat exchanger E-207A, leading to loop A RHR suppression pool cooling and drywell spray modes being unavailable for containment pressure reduction. Phase I SAMAs have already been implemented to use firewater for drywell spray and to use venting via DTV path to reduce containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
LCI-HTX-VF-E207B	3.24E-04	1.039	Loop A heat exchanger E-207B failure	This term represents random failure of RHR heat exchanger E-207B, leading to loop B RHR suppression pool cooling and drywell spray modes being unavailable for containment pressure reduction. Phase I SAMAs have already been implemented to use firewater for drywell spray and to use venting via DTV path to reduce containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated.
IE-T2	8.90E-02	1.038	Loss of PCS transients	This term represents an initiating event caused by a transient with PCS unavailable. Industry efforts over the last twenty years have led to a significant reduction of plant scrams from all causes. Phase II SAMA 038, to improve MSIV design and mitigate the consequences of this event, was evaluated.
RHR-MAI-MA-HTXBP	2.69E-04	1.032	RHR heat exchanger E-207B unavailable due to maintenance	This term represents RHR heat exchanger E-207B unavailable due to maintenance, leading to loop B RHR suppression pool cooling and drywell spray modes being unavailable for containment pressure reduction. Phase I SAMAs have already been implemented to use firewater for drywell spray and to use venting via DTV path to reduce containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
RBC-MAI-MA-LOOPB	2.36E-04	1.029	RBCCW loop B out for maintenance	This term represents RBCCW loop B unavailable due to maintenance. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
DWS-XHE-FO-W2	2.85E-04	1.026	Operator fails to align drywell spray mode of RHR	This term represents operator failure to align the drywell spray mode of RHR for containment pressure reduction. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
SPC-XHE-FO-W1	1.54E-04	1.026	Operator fails to align suppression pool cooling mode of RHR	This term represents operator failure to align the suppression pool cooling mode of RHR for containment pressure reduction. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
LCS-CCF-PG-STNRS	2.22E-05	1.024	Common cause failure of strainers BS-8002A&B plugged	This term represents common cause failure of the core spray and RHR suction strainers. A Phase I SAMA, installing improved passive emergency core cooling system (ECCS) suction strainers, has been implemented. Phase II SAMAs 042, 044, and 045, which recommend addition of independent injection systems to mitigate this failure event, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
DC1-CBR-CO-7216A	5.11E-05	1.023	125VDC circuit breaker 72-16A fails to remain closed	This term represents random failure of 125VDC circuit breaker 72-16A, leading to loss of DC power to bus D-16. Phase I SAMAs to improve battery charging capability and replace existing batteries with more reliable ones have already been installed. Phase II SAMAs 025, 026, 027, 031, 032, 033, 034, and 035 for enhancing DC system availability and reliability were evaluated.
ADS-XHE-FO-X1T2	6.88E-04	1.023	Operator fails to perform emergency depressurization (transient)	This term represents operator failure to manually open the SRVs for depressurization during transients. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
DC1-CBR-CO-72165	5.11E-05	1.023	125VDC circuit breaker 72-165 fails to remain closed	This term represents random failure of DC circuit breaker 72-165 to provide power to DTV valve AO 5025, causing failure of the valve to open on demand, resulting in loss of containment venting capability. Phase II SAMA 056 to improve DTV valve availability was evaluated.
OSP-SBO	7.64E-02	1.023	Operator fails to start or align station blackout (SBO) diesel to either bus A5 or A6	This term represents operator failure to start or align the SBO diesel to either bus A5 or A6 during a LOOP event. Phase I SAMAs, including improvement of SBO procedures and training to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
DC1-CBR-CO-7217A	5.11E-05	1.023	125VDC circuit breaker 72-17A fails to remain closed	This term represents random failure of 125VDC circuit breaker 72-17A, leading to loss of DC power to bus D-17. Phase I SAMAs to improve battery charging capability and replace existing batteries with more reliable ones have already been installed. Phase II SAMAs 025, 026, 027, 031, 032, 033, 034, and 035 for enhancing DC system availability and reliability were evaluated.
OSP-14	4.10E-02	1.022	Failure to recover offsite power within 14 hours	This term represents operator failure to recover offsite power within 14 hours during a LOOP event. Phase I SAMAs, including improvement of SBO procedures and training to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
IE-T3A	8.60E-01	1.022	Transients with condenser initially available	This term represents an initiating event caused by a transient with PCS available. Industry efforts over the last twenty years have led to a significant reduction of plant scrams from all causes. Phase II SAMA 038 to improve MSIV design and mitigate the consequences of this event was evaluated.
FXT-MAI-MA-P140	9.22E-03	1.019	Diesel driven fire water pump P-140 unavailable due to maintenance	This term represents diesel fire pump P-140 in maintenance. Phase II SAMA 045, to add a diverse injection system and provide an injection source other than fire water, was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
AC4-RCK-NO-604	2.51E-03	1.019	4.16kV circuit breaker 152-604 control circuit no output	This term represents failure of the control circuit of 4.16kV circuit breaker 152-604, leading to LOOP to safety bus A6. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedure to repair or replace failed 4.16kV breakers have already been installed. In addition, a Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
DC1-CBR-CO-72175	5.11E-05	1.018	125VDC circuit breaker 72-175 fails to remain closed	This term represents random failure of DC circuit breaker 72-175 to provide power to DTV valve AO 5042B, causing failure of the valve to open on demand, resulting in loss of containment venting capability. Phase II SAMA 056 to improve DTV valve availability was evaluated.
CIV-RCK-NO-5042B	2.50E-03	1.018	SV 5042B control circuit failure	This term represents random failure of the control circuit of DTV valve AO 5042B, causing failure of the valve to open on demand, resulting in loss of containment venting capability to control containment pressure. Phase II SAMA 056 to improve DTV valve availability was evaluated.
CIV-RCK-NO-A5025	2.50E-03	1.018	AO 5025 control circuit failure	This term represents random failure of the control circuit of DTV valve AO 5025, causing failure of the valve to open on demand, resulting in loss of containment venting capability to control containment pressure. Phase II SAMA 056 to improve DTV valve availability was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
AC4-RCK-NO-504	2.51E-03	1.017	4.16kV circuit breaker 152-504 control circuit no output	This term represents failure of the control circuit of 4.16kV circuit breaker 152-504, leading to LOOP to safety bus A5. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedures to repair or replace failed 4.16kV breakers have already been installed. In addition, a Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
SSW-MDP-FS-P208D	2.02E-03	1.017	SSW pump P-208D fails to start on demand	This term represents random failure of SSW pump P-208D to start. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.
SSW-CCF-FS-3P208	2.26E-05	1.017	Common cause failure of 3 SSW pumps to start	This term represents common cause failure of 3 service water pumps to start. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
SSW-MDP-FS-P208E	2.02E-03	1.016	SSW pump P-208E fails to start on demand	This term represents random failure of SSW pump P-208E to start. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.
IE-S1	3.00E-04	1.015	Medium LOCA	This term represents the medium LOCA initiating event. Several Phase I SAMAs have been implemented to provide more reliable or diverse high or low pressure injection systems to mitigate this event. Phase II SAMAs 040, 041, 042, 043, 044, and 054 were evaluated to reduce the CDF contribution from medium LOCA.
LCS-STR-PG-8002A	1.20E-04	1.014	ECCS strainer BS-8002A plugged	This term represents failure of core spray and RHR suction strainer BS-8002A. A Phase I SAMA was implemented to install improved passive ECCS suction strainers. Phase II SAMAs 042, 044, and 045, which recommend addition of independent injection systems to mitigate this failure event, were evaluated.
LCS-STR-PG-8002B	1.20E-04	1.014	ECCS strainer BS-8002B plugged	This term represents failure of core spray and RHR suction strainer BS-8002B. A Phase I SAMA was implemented to install improved passive ECCS suction strainers. Phase II SAMAs 042, 044, and 045, which recommend addition of independent injection systems to mitigate this failure event, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
ADS-XHE-FO-X1S1	7.40E-03	1.013	Operator fails to perform emergency depressurization during medium LOCA	This term represents operator failure to manually open the SRVs for depressurization during medium LOCA. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
EDG-ENG-FR-EDGB	6.10E-03	1.013	Emergency diesel generator -B (EDG) fails to continue to run	This term represents random failure of EDG-B, leading to an SBO event. Phase I SAMAs to improve availability and reliability of the EDGs by creating a cross-tie of EDGs fuel oil supply and installing a backup SBO diesel generator have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035, for enhancing AC or DC system reliability or to cope with LOOP and SBO events, were evaluated.
AC8-CBR-CO-104	9.50E-05	1.013	480V circuit breaker 52-104 fails to remain closed	This term represents random failure of 480V circuit breaker 52-104, leading to loss of power to 480V MCC B17 and its associated loads. A Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 030 and 058 to improve 480V bus availability were evaluated.
HCI-MAI-MA-HCITM	1.62E-02	1.013	HPCI unavailable due to maintenance	This term represents HPCI system unavailable due to maintenance. Phase I SAMAs to improve availability and reliability of the HPCI system that have already been implemented include raising backpressure trip setpoints and proceduralizing intermittent operation. Additional improvements were evaluated in Phase II SAMAs 040, 041, 042, 043, 044, and 045.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
SSW-CCF-FR-3P208	5.59E-06	1.012	Common cause failure of 3 SSW pumps to run	This term represents common cause failure of 3 service water pumps to continue to run Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.
AC8-CBR-CO-205	9.50E-05	1.012	480V circuit breaker 52-205 fails to remain closed	This term represents random failure of 480V circuit breaker 52-205, leading to loss of power to 480V MCC B18 and its associated loads. A Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 030 and 058 to improve 480V bus availability were evaluated.
IE-T3C	4.40E-02	1.012	Inadvertently opened relief valve	This term represents an initiating event caused by inadvertent opening of a relief valve. Improvement of the SRV design and SRV reseat reliability, to reduce the probability and consequences of this initiating event, were evaluated in Phase II SAMAs 046 and 050.
RBC-CCF-CC-4MOVS	1.13E-05	1.012	Common cause failure of RBCCW heat exchanger A & B side MOVs (4) to open	This term represents common cause failure of RBCCW heat exchanger A & B side MOVs to open. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
OSP-24	1.41E-02	1.011	Failure to recover offsite power within 24 hours	This term represents operator failure to recover offsite power within 24 hours during a LOOP event. Phase I SAMAs, including improvement of SBO procedures and training to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.
SSW-RCI-FE-3828X	3.00E-04	1.01	Pressure switch PS-3828X coil fails to energize	This term represents random failure of SSW pressure switch PS-3828X, resulting in loss of SSW system loop A. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.
EDG-MAI-MA-EDGA	6.41E-03	1.01	EDG-A out for maintenance	This term represents EDG-A out for maintenance, leading to an SBO event. Phase I SAMAs to improve availability and reliability of the EDGs by creating a cross-tie of EDGs fuel oil supply and installing a backup SBO diesel generator have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035, for enhancing AC or DC system reliability or to cope with LOOP and SBO events, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
EDG-ENG-FR-EDGA	6.10E-03	1.01	EDG-A fails to continue to run	This term represents random failure of EDG-A, leading to an SBO event. Phase I SAMAs to improve availability and reliability of the EDGs by creating a cross-tie of EDGs fuel oil supply and installing a backup SBO diesel generator have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035, for enhancing AC or DC system reliability or to cope with LOOP and SBO events, were evaluated.
SSW-MOV-OO-V3805	6.62E-04	1.009	SSW TBCCW A heat exchanger outlet MOV MO-3805 fails to go 90% closed	This term represents random failure of SSW MOV MO-3805 to go 90% closed, resulting in loss of SSW to RBCCW loop B. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
SSW-MDP-FS-P208B	2.02E-03	1.009	SSW pump P-208B fails to start on demand	This term represents random failure of SSW pump P-208B to start. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.
SSW-MDP-FS-P208A	2.02E-03	1.009	SSW pump P-208A fails to start on demand	This term represents random failure of SSW pump P-208A to start. Phase I SAMAs were implemented to improve service water system reliability by enhancing screen wash, adding redundant DC control power for SSW pumps, and increasing seismic integrity of the partition wall between the SSW pumps. Phase II SAMA 055 to improve SSW system reliability by reducing common dependencies was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
C	5.80E-06	1.009	Reactor Protection System (RPS) failure	This term represents failure of the RPS. Several Phase I SAMAs to minimize the risks associated with anticipated transient without scram (ATWS) scenarios have already been installed. No Phase II SAMAs were evaluated to further improve reliability of RPS. However, Phase II SAMA 048 to enhance reliability of the standby liquid control system and improve capability to mitigate the consequences of an ATWS event was evaluated.
AC4-RCK-NO-605	2.51E-03	1.009	4.16kV circuit breaker 152-605 control circuit no output	This term represents failure of the control circuit of 4.16kV circuit breaker 152-605, leading to loss of power to safety bus A6. Phase I SAMAs to improve 4.16kV bus cross-tie capability and procedures to repair or replace failed 4.16kV breakers have already been installed. In addition, a Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
RCI-TDP-RS-P206	1.52E-02	1.009	RCIC turbine driven pump P-206 fails to restart after clear high level signal	This term represents random failure of the RCIC system. Phase I SAMAs to improve availability and reliability of the RCIC system that have already been implemented include raising backpressure trip setpoints and proceduralizing intermittent operation. Additional improvements were evaluated in Phase II SAMAs 040, 041, 042, 043, 044, and 045.
FXT-RCK-NO-P140	2.50E-03	1.009	Diesel fire pump P-140 control circuit no output	This term represents diesel fire pump P-140 control circuit failure. Phase II SAMA 045, to add a diverse injection system and provide an injection source other than fire water, was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
AC4-RCK-NO-508	2.51E-03	1.008	4.16kV circuit breaker 152-508 control circuit no output	This term represents failure of the control circuit of 4.16kV circuit breaker 152-508, leading to loss of power to 480V load center B1. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedures to repair or replace failed 4.16kV breakers have already been implemented. In addition, a Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
AC8-RCK-NO-101	2.50E-03	1.008	480V circuit breaker 52-101 control circuit no output	This term represents random failure of 480V circuit breaker 52-101, leading to loss of power to 480V load center B1 and its associated loads. A Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 030 and 058 to improve 480V bus availability were evaluated.
EDG-MAI-MA-EDGB	4.09E-03	1.008	EDG-B out for maintenance	This term represents EDG-B out for maintenance, leading to an SBO event. Phase I SAMAs to improve availability and reliability of the EDGs by creating a cross-tie of EDGs fuel oil supply and installing a backup SBO diesel generator have already been implemented. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035, for enhancing AC or DC system reliability or to cope with LOOP and SBO events, were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
HCI-TDP-FS-PM205	7.53E-03	1.008	HPCI turbine driven pump P-205 fails to start on demand	This term represents random failure of the HPCI system. Phase I SAMAs to improve availability and reliability of the HPCI system that have already been implemented include raising backpressure trip setpoints and proceduralizing intermittent operation. Additional improvements were evaluated in Phase II SAMAs 040, 041, 042, 043, 044, and 045.
RBC-CCF-FS-4PUMP	7.35E-06	1.008	Common cause failure of four RBCCW pumps to start	This term represents common cause failure of four RBCCW pumps to start. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
AC4-RCK-NO-505	2.51E-03	1.007	4.16kV circuit breaker 152-505 control circuit no output	This term represents failure of the control circuit of 4.16kV circuit breaker 152-505, leading to loss of power supply to safety bus A5. Phase I SAMAs to improve 4.16kV bus cross-tie capability and revise procedures to repair or replace failed 4.16kV breakers have already been installed. In addition, a Phase I SAMA was implemented to proceduralize operator action to manually close the circuit breaker. Phase II SAMAs 025, 026, 027, 028, 029, 030, 033, and 035 for enhancing AC or DC system reliability or to cope with LOOP and SBO events were evaluated.
FXT-XVM-CC-511	5.00E-04	1.007	Manual valve 10-HO-511 fails to open	This term represents random failure of manual valve 10-HO-511 to open to provide fire water to LPCI loops A and B. This failure leads to loss of fire water backup for reactor vessel injection and drywell spray. Phase II SAMA 059 to enhance availability of the fire water system was evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
FXT-XVM-CC-8I56	5.00E-04	1.007	Manual valve 8-I-56 fails to open	This term represents random failure of manual valve 8-I-56 to open to provide fire water to LPCI loops A and B. This failure leads to loss of fire water backup for reactor vessel injection and drywell spray. Phase II SAMA 059 to enhance availability of the fire water system was evaluated.
RCI-MAI-MA-RCITM	1.97E-02	1.007	RCIC unavailable due to maintenance	This term represents RCIC system unavailable due to maintenance. Phase I SAMAs to improve availability and reliability of the RCIC system that have already been implemented include raising backpressure trip setpoints and proceduralizing intermittent operation. Additional improvements were evaluated in Phase II SAMAs 040, 041, 042, 043, 044, and 045.
CIV-AOV-CC-5042B	1.00E-03	1.007	AO 5042B fails to open on demand	This term represents random failure of DTV valve AO 5042B to open on demand, resulting in loss of containment venting capability to control containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated for containment pressure control.
CIV-AOV-CC-A5025	1.00E-03	1.007	AO 5025 fails to open on demand	This term represents random failure of DTV valve AO 5025 to open on demand, resulting in loss of containment venting capability to control containment pressure. Phase II SAMAs 001, 009, 014, and 059, to provide alternate means of suppression pool cooling and drywell spray and to enhance the availability and reliability of firewater for reactor vessel injection and drywell spray, were evaluated for containment pressure control.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
CM	3.30E-01	1.006	RPS mechanical failure	This term represents random failure of the RPS. Several Phase I SAMAs to minimize the risks associated ATWS scenarios have already been installed. No Phase II SAMAs were evaluated to further improve reliability of RPS. However, Phase II SAMA 048 to enhance reliability of the standby liquid control system and improve ATWS capability to mitigate the consequences of this event was evaluated.
RBC-MAI-MA-P202E	6.71E-03	1.006	RBCCW pump 202E out for maintenance	This term represents RBCCW pump 202E unavailable due to maintenance. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
RBC-MAI-MA-P202F	6.44E-03	1.006	RBCCW pump 202F out for maintenance	This term represents RBCCW pump 202F unavailable due to maintenance. A Phase I SAMA was implemented to improve RBCCW system reliability by making component cooling water trains separate. Phase II SAMA 055 to improve RBCCW system reliability by reducing common dependencies was evaluated.
IE-TDC-CCF	3.66E-08	1.006	Common cause failure of 125VDC buses A&B	This term represents an initiating event caused by a complete loss of 125VDC buses D-16 and D-17 or random failure of batteries D-1 and D-2. Phase I SAMAs to improve battery charging capability and replace existing batteries with more reliable ones have already been installed. Phase II SAMAs 025, 026, 027, 031, 032, 033, 034, and 035 for enhancing DC system availability and reliability were evaluated.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
SPC-MAI-MA-SPCA	3.01E-03	1.005	Suppression pool cooling loop A out for maintenance	This term represents RHR suppression pool cooling loop A unavailable due to maintenance. Phase I SAMAs to improve availability and reliability of the RHR suppression pool cooling mode that have already been implemented include using drywell spray mode and fire protection cross-tie to provide redundant containment heat removal capability. Additional improvements were evaluated in Phase II SAMAs 001 and 014.
SPC-MAI-MA-SPCB	2.91E-03	1.005	Suppression pool cooling loop B out for maintenance	This term represents RHR suppression pool cooling loop B unavailable due to maintenance. Phase I SAMAs to improve availability and reliability of the RHR suppression pool cooling mode that have already been implemented include using drywell spray mode and fire protection cross-tie to provide redundant containment heat removal capability. Additional improvements were evaluated in Phase II SAMAs 001 and 014.
DWS-MAI-MA-DWSA	3.18E-03	1.005	Drywell spray loop A out for maintenance	This term represents RHR drywell spray loop A unavailable due to maintenance. Phase I SAMAs to improve availability and reliability of the RHR drywell spray mode that have already been implemented include using suppression pool cooling mode and fire protection cross-tie to provide redundant containment heat removal capability. Additional improvements were evaluated in Phase II SAMA 009.

Table E.1-3
Correlation of Level 1 Risk Significant Terms to Evaluated SAMAs

Event Name	Probability	RRW	Event Description	Disposition
ADS-XHE-FO-X1S2	1.45E-03	1.005	Operator fails to perform emergency depressurization during small LOCA	This term represents operator failure to manually open the SRVs for depressurization during a small LOCA. Phase I SAMAs, including improvement of procedures and installation of instrumentation to enhance the likelihood of success of operator action in response to accident conditions, have already been implemented. No additional Phase II SAMAs were recommended for this subject.

E.1.2 PSA Model – Level 2 Analysis

E.1.2.1 Containment Performance Analysis

The PNPS Level 2 PSA model used for the SAMA analysis is the most recent internal events risk model, which is an updated version of the model used in the IPE [References E.1-2 and E.1-3]. The Level 2 PSA model used for the SAMA analysis, Revision 1, reflects the PNPS operating configuration and design changes as of September 2001. Specifically, the Level 2 model has been updated to incorporate insights from the independent BWROG peer review.

The PNPS Level 2 model includes two types of considerations: (1) a deterministic analysis of the physical processes for a spectrum of severe accident progressions, and (2) a probabilistic analysis component in which the likelihood of the various outcomes are assessed. The deterministic analysis examines the response of the containment to the physical processes during a severe accident. This response is performed by

- utilization of the MAAP code [Reference E.1-4] to simulate severe accidents that have been identified as dominant contributors to core damage in the Level 1 analysis, and
- reference calculation of several hydrodynamic and heat transfer phenomena that occur during the progression of severe accidents. Examples include debris coolability, pressure spikes due to ex-vessel steam explosions, scoping calculation of direct containment heating, molten debris filling the pedestal sump and flowing over the drywell floor, containment bypass, deflagration and detonation of hydrogen, thrust forces at reactor vessel failure, liner melt-through, and thermal attack of containment penetrations.

The Level 2 analysis examined the dominant accident sequences and the resulting plant damage states (PDS) defined in Level 1. The Level 1 analysis involves the assessment of those scenarios that could lead to core damage. A list of the PDS groups and descriptions from the Level 2 analysis is presented in Table E.1-4.

A full Level 2 model was developed for the IPE and completed at the same time as the Level 1 model. The Level 2 model consists of a single containment event tree (CET) with functional nodes that represent phenomenological events and containment protection system status. The nodes were quantified using subordinate trees and logic rules. A list of the CET functional nodes and descriptions used for the Level 2 analysis is presented in Table E.1-5.

The Large Early Release Frequency (LERF) is an indicator of containment performance from the Level 2 results because the magnitude and timing of these releases provide the greatest potential for early health effects to the public. The frequency calculated is approximately

Table E.1-4
Summary of PNPS PSA Core Damage Accident Class

PDS Group	Simplified Description	Point Estimate	% of Total CDF
LOCAs	Large and small break LOCA with initial or long-term loss of core cooling. Core damage results at low or high reactor pressure. For most PDS, late injection and containment heat removal are available.	1.16E-7	1.80
TRANS	Short and long-term transient events. Core damage results at either low or high reactor pressure. Late injection and containment heat removal are available.	2.43E-7	3.79
SBO	SBO involving a loss of high-pressure injection. Core damage results at either low (stuck-open SRV) or high reactor pressure. All accident mitigating functions are recoverable when AC power is restored.	1.48E-7	2.31
VSL_RUPT	Vessel rupture event resulting in LOCA beyond ECCS capability. All PDS result in core damage at low reactor pressure with late injection available.	4.00E-9	0.06
ATWS	Short-term ATWS that leads to early core damage at high reactor pressure following loss of reactivity control and rapid containment pressurization. Reactor coolant system leakage rates associated with boil-off of coolant through the cycling of SRVs/SV with early core melt subsequent to containment overpressure failure. Late injection and containment heat removal are available.	3.39E-8	0.53
ISLOCA	Large and small break interfacing system LOCA outside containment. Core damage results at low or high reactor pressure with a bypassed containment.	4.00E-9	0.06
TW	Containment decay heat removal systems are not available and coolant recirculation to the torus over pressurizes the containment to failure or venting. The torus is saturated.	5.86E-6	91.45
	Total	**6.41E-06**	**1.00E+00**

Table E.1-5
Notation and Definitions for PNPS CET Functional Nodes Description

CET Node	CET Functional Node Description
Plant Damage State Event (PDS_EVNT)	This top event represents the initiators considered in the containment performance analysis.
RPV Pressure at Vessel Failure (RPV@VF)	This top event identifies the status of the reactor pressure vessel (RPV) pressure. RPV@VF is set to success when RPV pressure is low. RPV@VF is set to failure when RPV is high.
In-Vessel Cooling Recovery (IN-REC)	This top event addresses the recovery of coolant injection into the vessel after core degradation, but prior to vessel breach. This top event considers the possibility of low-pressure injection systems working once the RPV is depressurized.
Vessel Failure (VF)	This top event addresses recovery from core degradation within the vessel and the prevention of vessel head thermal attack. Core melt recovery requires the recovery of core cooling prior to core blocking or relocation of molten debris to the lower plenum and thermal attack of the vessel head.
Early Containment Failure (CFE)	This top event node considers the potential loss of containment integrity at, or before, vessel failure. Several phenomena are considered credible mechanisms for early containment failure. They may occur alone or in combination. The phenomena are containment isolation failure; containment bypass; containment overpressure failure at vessel breach; hydrogen deflagration or detonation; fuel-coolant interactions (steam explosions); high pressure melt ejection and subsequent direct containment heating; and drywell steel shell melt-through.
Early Release to Torus (EPOOL)	This top event node considers the importance of early torus pool scrubbing in mitigating the magnitude of fission products released from the damaged core. Success implies that fission product transport path subsequent to early containment failure is through the torus water and the torus airspace. That is, the torus pool is not bypassed. Failure involves a release into the drywell.
Debris Cooled Ex-vessel (DCOOL)	This top event considers the delivery of water to the drywell, via drywell sprays, or via injection to the RPV and drainage out an RPV breach onto the drywell floor. Success implies the availability of water and the formation of a coolable debris bed such that concrete attack is precluded. Failure implies that the molten core attacks concrete in the reactor pedestal, that core debris remains hot, and sparing of the concrete decomposition products through the melt releases the less volatile fission products to the containment atmosphere.

Table E.1-5
Notation and Definitions for PNPS CET Functional Nodes Description (Continued)

CET Node	CET Functional Node Description
Late Containment Failure (CFL)	This top event addresses the potential loss of containment integrity in the long-term. Late containment failure may result from long-term steam and non-condensable gas generation from the attack of molten core debris on concrete.
Late Release to Torus (LPOOL)	This top event node considers the importance of late torus pool scrubbing in mitigating the magnitude of fission products released from the damaged core. Success implies that fission product transport path subsequent to late containment failure is through the torus water and the torus airspace. That is, the torus pool is not bypassed. Failure involves a release into the drywell.
Fission Product Removal (FPR)	This top event addresses fission product releases from the fuel into the containment and airborne fission product removal mechanisms within the containment structure to characterize potential magnitude of fission product releases to the environment should the containment fail. Failure implies that most of the fission products from the fuel and containment are ultimately released to the environment without mitigation.
Reactor Building (RB)	This top event is used to assess the ability of the reactor building to retain fission products released from containment. Success of top event RB is defined to be a reduction of the containment release magnitude.

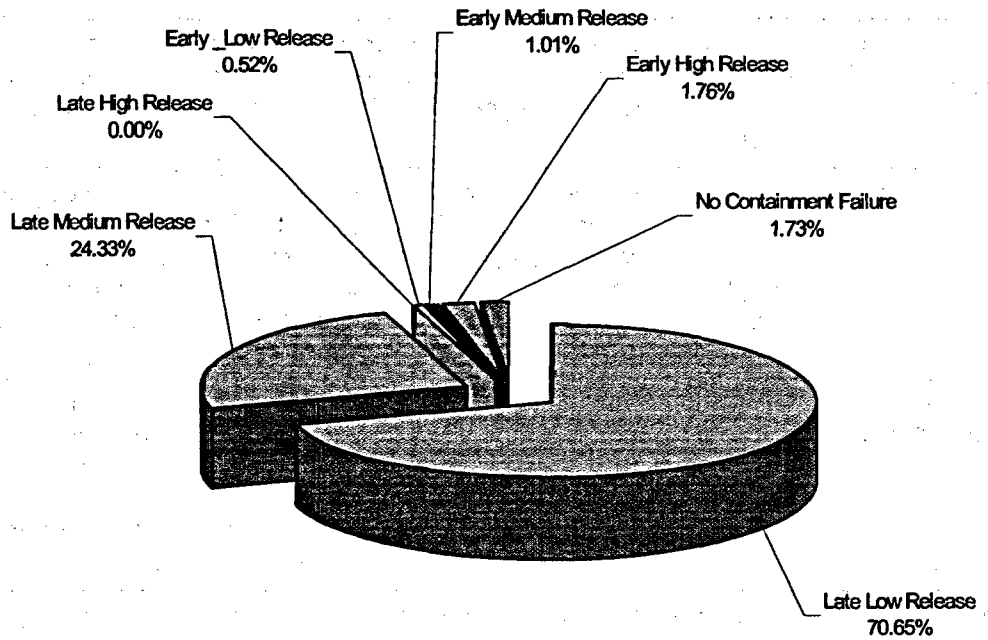

Figure E.1-1
PNPS Radionuclide Release Category Summary

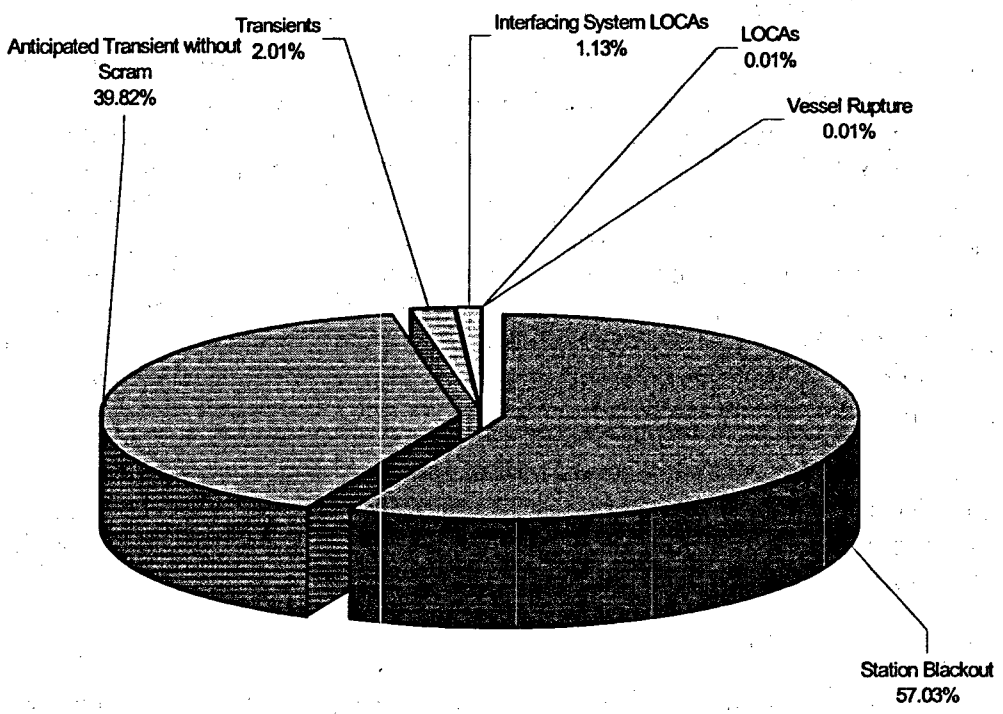

Figure E.1-2
PNPS Plant Damage State Contribution to LERF

E.1.2.2 Radionuclide Analysis

E.1.2.2.1 Introduction

A major feature of a Level 2 analysis is the estimation of the source term for every possible outcome of the CET. The CET end points represent the outcomes of possible in-containment accident progression sequences. These end points represent complete severe accident sequences from initiating event to release of radionuclides to the environment. The Level 1 and plant system information is passed through to the CET evaluation in discrete PDS. An atmospheric source term may be associated with each of these CET sequences. Because of the large number of postulated accident scenarios considered, mechanistic calculations (i.e., MAAP calculations) are not performed for every end-state in the CET. Rather, accident sequences produced by the CET are grouped or "binned" into a limited number of release categories each of which represents all postulated accident scenarios that would produce a similar fission product source term.

The criteria used to characterize the release are the estimated magnitude of total release and the timing of the first significant release of radionuclides. The predicted source term associated with each release category, including both the timing and magnitude of the release, is determined using the results of MAAP calculations [Reference E.1-4].

E.1.2.2.2 Timing of Release

Timing completely governs the extent of radioactive decay of short-lived radioisotopes prior to an off-site release and, therefore, has a first-order influence on immediate health effects. PNPS characterizes the release timing relative to the time at which the release begins, measured from the time of accident initiation. Two timing categories are used: early (0-24 hours) and late (>24 hours).

Based on MAAP calculations for a spectrum of severe accident sequences, PNPS expects that an Emergency Action Level (as defined by the PNPS Emergency Plan) will be reached within the first half hour after accident initiation. Reaching an Emergency Action Level initiates a formal decision-making process that is designed to provide public protective actions. Within 24 hours of accident initiation, the Level 2 analysis assumed that off-site protective measures would be effective. Therefore, the definitions of the release timing categories are as follows.

- Early releases are CET end-states involving containment failure prior to or at vessel failure or after vessel failure and occurring within 0 to 24 hours measured from the time of accident initiation and for which minimal offsite protective measures would be accomplished.

- Late releases are CET end-states involving containment failure greater than 24 hours from the time of accident initiation, for which offsite measures are fully effective.

E.1.2.2.3 Magnitude of Release

Source term results from previous risk studies suggest that categorization of release magnitude based on cesium iodide (CsI) release fractions alone are appropriate [Reference E.1-5]. The CsI release fraction indicates the fraction of in-vessel radionuclides escaping to the environment. (Noble gas release levels are non-informative since release of the total core inventory of noble gases is essentially complete given containment failure).

The source terms were grouped into four distinct radionuclide release categories or bins according to release magnitude as follows:

(1) High (HI) - A radionuclide release of sufficient magnitude to have the potential to cause early fatalities. This implies a total integrated release of >10% of the initial core inventory of CsI [Reference E.1-5].[1]

(2) Medium (MED) - A radionuclide release of sufficient magnitude to cause near-term health effects. This implies a total integrated release of between 1 and 10% of the initial core inventory of CsI [Reference E.1-5].[2]

(3) Low (LO) - A radionuclide release with the potential for latent health effects. This implies a total integrated release of between 0.001% and 1% of the initial core inventory of CsI.

(4) Negligible (NCF) - A radionuclide release that is less than or equal to the containment design base leakage. This implies total integrated release of <0.001% of the initial core inventory of CsI.

The "total integrated release" as used in the above categories is defined as the integrated release within 36 hours after RPV failure. If no RPV failure occurs, then the "total integrated release" is defined as the integrated release within 36 hours after accident initiation.

E.1.2.2.4 Release Category Bin Assignments

Table E.1-6 summarizes the scheme used to bin sequences with respect to magnitude of release, based on the predicted CsI release fraction and release timing. The combination of release magnitude and timing produce seven distinct release categories for source terms. These are the representative release categories presented in Table E.1-7.

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1. Once the CsI source term exceeds 0.1, the source term is large enough that doses above the early fatality threshold can sometimes occur within a population center a few miles from the site.
2. The reference document indicates that for CsI release fractions of 1 to 10%, the number of latent fatalities is found to be at least 10% of the latent fatalities for the highest release.

Table E.1-6
Release Severity and Timing Classification Scheme Summary

Release Severity		Release Timing	
Classification Category	**CsI % Release**	**Classification Category**	**Time of Initial Release from Accident Initiation**
High	Greater than 10	Early (E)	Less than 24 hours
Medium	1 to 10		
Low	0.001 to 1	Late (L)	Greater than 24 hours
Negligible	Less than 0.001		

Table E.1-7
PNPS Release Categories

Timing of Release	Magnitude of Release			
	Low	Medium	High	
Early	Early/Low	Early/Med	Early/High	NCF
Late	Late/Low	Late/Med	Late/High	

E.1.2.2.5 Mapping of Level 1 Results into the Various Release Categories

PDS provide the interface between the Level 1 and Level 2 analyses (i.e. between core damage accident sequences and fission product release categories). In the PDS analysis, Level 1 results were grouped ("binned") according to plant characteristics that define the status of the reactor, containment, and core cooling systems at the time of core damage. This ensures that systems important to core damage in the Level 1 event trees, and the dependencies between containment and other systems are handled consistently in the Level 2 analysis. A PDS therefore represents a grouping of Level 1 sequences that defines a unique set of initial conditions that are likely to yield a similar accident progression through the Level 2 CETs and the attendant challenges to containment integrity.

From the perspective of the Level 2 assessment, PDS binning entails the transfer of specific information from the Level 1 to the Level 2 analyses.

- *Equipment failures in Level 1.* Equipment failures in support systems, accident prevention systems, and mitigation systems that have been noted in the Level 1 analysis are carried into the Level 2 analysis. In this latter analysis, the repair or recovery of failed equipment is not allowed unless an explicit evaluation, including a consideration of

adverse environments where appropriate, has been performed as part of the Level 2 analysis.

- *RPV status.* The RPV pressure condition is explicitly transferred from the Level 1 analysis to the CET.
- *Containment status.* The containment status is explicitly transferred from the Level 1 analysis to the CET. This includes recognition of whether the containment is bypassed or is intact at the onset of core damage.
- *Accident sequence timing.* Differences in accident sequence timing are transferred with the Level 1 sequences. Timing affects such sequences as SBO, internal flooding, and containment bypass (ISLOCA).

This transfer of information allows timing to be properly assessed in the Level 2 analysis.

Based on the above criteria, the Level 1 results were binned into 48 PDS. These PDS define important combinations of system states that can result in distinctly different accident progression pathways and, therefore, different containment failure and source term characteristics. Table E.1-8 provides a description of the PNPS PDS that are used to summarize the Level 1 results.

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States

PDS	Description	Point Estimate	% of CDF
PDS-1	Long-term LOCA with loss of high-pressure core makeup from HPCI and RCIC, loss of containment heat removal, and failure to depressurize the primary system for low-pressure core makeup. Core damage results at high primary system pressure. Late injection from low-pressure systems (core spray, LPCI, and firewater) is available, provided primary system depressurization occurs. The containment is vented and intact.	0.00E+00	0.00
PDS-2	Long-term LOCA with loss of both high-pressure core makeup (HPCI and RCIC) and containment heat removal. Core damage results at high primary system pressure. Because containment venting fails, containment failure occurs long-term. Late injection is available from low-pressure systems (core spray, LPCI, and fire water) provided they survive containment failure.	1.05E-11	<0.001

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-3	Short-term LOCA with loss of high-pressure core makeup and failure to depressurize the primary system for low-pressure core makeup. Core damage occurs at high primary system pressure. Late injection from core spray, LPCI, and firewater is available, provided primary system depressurization occurs. Containment heat removal is available.	8.68E-08	1.35
PDS-4	Short-term LOCA with loss of high-pressure core makeup, loss of containment heat removal, and failure to depressurize the primary system for low-pressure core makeup. Core damage occurs at high primary system pressure. Late injection from core spray, LPCI, and firewater is available, provided primary system depressurization occurs. Unlike PDS-3, containment heat removal is unavailable.	0.00E+00	<0.001
PDS-5	Long-term LOCA with loss of high-pressure core makeup and containment heat removal. Core damage occurs at low primary system. Late injection is available from low-pressure systems (core spray, LPCI, and fire water). The containment is vented and intact.	0.00E+00	0.00
PDS-6	Long-term large LOCA. High-pressure core makeup from HPCI and RCIC are unavailable due to the large LOCA. Because containment venting fails, containment failure occurs long-term. Late injection is available from low-pressure systems (core spray, LPCI, and fire water) provided they survive containment failure. Core damage occurs at low primary system pressure.	0.00E+00	0.00
PDS-7	Short-term large LOCA with loss of core cooling. Core damage results at low primary system pressure. Late injection from firewater cross tie and containment heat removal are available.	1.12E-09	0.08
PDS-8	Short-term large LOCA with loss of core cooling. Core damage results at low primary system pressure. Late injection from firewater cross tie is available. However, unlike PDS-7, containment heat removal is unavailable.	4.43E-09	0.07

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-9	Short-term LOCA with loss of high and low-pressure core cooling. Because the primary system is depressurized, core damage results at low primary system pressure. Late injection from SSW system, containment venting, and containment heat removal are available.	3.64E-09	0.06%
PDS-10	Short-term LOCA with loss of high and low-pressure core cooling. Because the primary system is depressurized, core damage results at low primary system pressure. Late injection from SSW system and containment heat removal are available. However, unlike PDS-9, containment venting is not available.	0.00E+00	0.00
PDS-11	Short-term LOCA with loss of high and low-pressure core cooling. Core damage results at low primary system pressure. Late injection from SSW system is available. However, unlike PDS-9, containment venting and containment heat removal are unavailable.	0.00E+00	0.00
PDS-12	Transient with a loss of long-term decay heat removal. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. The containment is vented and remains intact at the time of core damage.	2.37E-08	0.37
PDS-13	Transient with a loss of long-term decay heat removal. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. Unlike PDS-12 containment venting fails.	3.75E-06	58.5
PDS-14	Short-term transient with failure to depressurize the primary system. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. Containment heat removal from RHR is available.	1.52E-07	2.37
PDS-15	Short-term transient with failure to depressurize the primary system. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. Containment heat removal from RHR is available. However, containment venting is not available.	5.07E-08	0.79

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-16	Short-term transient with failure to depressurize the primary system. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. Containment heat removal from RHR is not available, but containment venting is available.	4.89E-09	0.08
PDS-17	Short-term transient with failure to depressurize the primary system. Core damage results at high primary system pressure. Late in-vessel and ex-vessel injection is available. Neither containment heat removal from RHR nor containment venting is available.	2.53E-09	0.04
PDS-18	Transient with a loss of long-term decay heat removal. Core damage results at low primary system pressure. Late in-vessel and ex-vessel injection is available. The containment is vented and remains intact at the time of core damage.	1.56E-06	24.40
PDS-19	Transient with a loss of long-term decay heat removal. Core damage results at low primary system pressure. Late in-vessel and ex-vessel injection is available. Unlike PDS-18 containment venting fails.	5.24E-07	8.18
PDS-20	Long-term transients with loss of core cooling. Core damage results at low primary system pressure. No late injection, but containment heat removal is available.	6.78E-11	0.001
PDS-21	Short-term transients (IORV) with loss of core cooling. Core damage results at low primary system pressure. Late injection and containment heat removal are available.	8.18E-09	0.13
PDS-22	Short-term transients with loss of core cooling. Core damage results at low primary system pressure. Late injection and containment heat removal are available. However, containment venting is not available.	1.08E-09	0.02
PDS-23	Short-term transients with loss of core cooling. Core damage results at low primary system pressure. Late injection and containment venting are available, but containment heat removal is not available.	0.00E+00	0.00
PDS-24	Similar to PDS-23, except that containment venting is not available.	4.98E-09	0.08

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-25	Short-term transients with loss of core cooling. Core damage results at low primary system pressure. No late injection, but containment heat removal and containment venting are available.	2.57E-09	0.04
PDS-26	Similar to PDS-25, except that containment venting is not available.	1.24E-08	0.19
PDS-27	Short-term transients with loss of core cooling. Core damage results at low primary system pressure. Late injection and containment heat removal are not available. However, containment venting is available	4.40E-11	0.001
PDS-28	Short-term transients with loss of core cooling. Core damage results at low primary system pressure. Late injection, containment heat removal and containment venting are not available.	1.10E-09	0.02
PDS-29	Long-term SBO involving loss of injection at high primary system pressure from battery depletion. All accident-mitigating functions are recoverable when AC power is restored.	1.41E-07	2.21
PDS-30	Short-term SBO sequence involving a loss of high-pressure injection at high primary system pressure from loss of all AC power and DC power or failure of SRVs. All accident-mitigating functions are recoverable when offsite power is restored.	0.00E+00	0.00
PDS-31	Long-term SBO sequence involving a loss of high-pressure injection due to one stuck-open safety relief valve or long-term failure of HPCI and RCIC and subsequent failure to depressurize the primary system. Core damage results at low primary system pressure. All accident-mitigating functions are recoverable when offsite power is restored.	2.60E-09	0.04
PDS-32	Short-term SBO sequence involving a loss of high-pressure injection due to two stuck-open safety relief valves or failure of HPCI and RCIC and one stuck-open safety relief valve. Core damage results at low primary system pressure. All accident-mitigating functions are recoverable when offsite power is restored.	4.00E-09	0.06

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-33	Short-term large reactor vessel rupture. The resulting loss of coolant is beyond the makeup capability of ECCS. Core damage occurs in the short term at low primary system pressure. Vessel injection and all forms of containment heat removal (RHR and containment venting) are available. The containment is not bypassed and AC power is available.	4.00E-09	0.06
PDS-34	Similar to PDS-33, except that containment heat removal from RHR fails.	0.00E+00	0.00
PDS-35	Short-term large reactor vessel rupture. The resulting loss of coolant is beyond the makeup capability of ECCS. Core damage occurs in the short term at low primary system pressure. Vessel injection is unavailable. However, all forms of containment heat removal (RHR and containment venting) are available. The containment is not bypassed and AC power is available.	0.00E+00	0.00
PDS-36	Similar to PDS-35, except that containment heat removal from RHR fails.	0.00E+00	0.00
PDS-37	Short-term ATWS with failure of SRVs and SVs to open to reduce primary system pressure. The ensuing primary system over pressurization leads to a LOCA beyond core cooling capabilities. Late injection and containment heat removal are available.	1.95E-08	0.31
PDS-38	Short-term ATWS that leads to early core damage at low primary system pressure following successful reactivity control. Late injection is not available. However, containment heat removal is available.	0.00E+00	0.00
PDS-39	Similar to PDS-38 except that containment heat removal from the RHR system is not available.	2.32E-09	0.04
PDS-40	Long-term ATWS that leads to late core damage at low primary system pressure following successful reactivity control. Late injection is available; containment heat removal from the RHR is not available. The containment is vented.	0.00E+00	0.00

Table E.1-8
Summary of PNPS Core Damage Accident Sequences Plant Damage States (Continued)

PDS	Description	Point Estimate	% of CDF
PDS-41	Short-term ATWS that leads to early core damage at high primary system pressure following successful reactivity control. Late injection and containment heat removal are available.	1.34E-11	<0.001
PDS-42	Similar to PDS-41 except that containment heat removal from the RHR system is not available.	0.00E+00	0.00
PDS-43	Long-term ATWS that leads to late core damage at high primary system pressure following successful reactivity control. Late injection is available; containment heat removal from the RHR is not available. The containment is vented.	0.00E+00	0.00
PDS-44	Long-term ATWS that leads to late core damage at high primary system pressure following successful reactivity control. Late injection is available. However, containment heat removal from the RHR system and containment venting are not available.	0.00E+00	0.00
PDS-45	Short-term ATWS that leads to containment failure and early core damage at high primary system pressure because of inadequate reactor water level following a loss of reactivity control. Late injection and containment venting are available.	3.39E-08	0.53
PDS-46	Short-term ATWS that leads to containment failure and early core damage at high primary system pressure because of inadequate reactor water level following successful reactivity control. No late injection; however, containment venting is available.	0.00E+00	0.00
PDS-47	Unisolated LOCA outside containment with early core melt at high RPV pressure.	3.22E-09	0.05
PDS-48	Unisolated LOCA outside containment with early core melt at low RPV pressure.	7.73E-10	0.01

The PDS designators listed in Table E.1-8 represent the core damage end state categories from the Level 1 analysis that are grouped together as entry conditions for the Level 2 analysis. The Level 2 accident progression for each of the PDS is then evaluated using a single CET to determine the appropriate release category for each Level 2 sequence. Each end state associated with a Level 2 sequence is assigned to one of the release categories depicted in Table E.1-7. Note, however, that since not all the Level 2 sequences associated with each Level 1 core damage class may be assigned to the same release category, there is no direct link between a specific Level 1 core damage PDS and Level 2 release category. Rather, the sum of the Level 2 end state frequencies assigned to each release category determines the overall frequency of that release category. The CET described in the Level 2 model determines the release category frequency attributed to each Level 1 core damage PDS.

E.1.2.2.6 Collapsed Accident Progression Bins Source Terms

The source term analysis results in hundreds of source terms for internal initiators, making calculation with the MACCS2 consequence model cumbersome. Therefore, the source terms were grouped into a much smaller number of source term groups defined in terms of similar properties, with a frequency weighted mean source term for each group.

The consequence analysis source terms groups are represented by collapsed accident progression bins (CAPB). The CAPB were generated by sorting the accident progression bins for each of the forty-eight PDS on attributes of the accident: the occurrence of core damage, the occurrence of vessel breach, primary system pressure at vessel breach, the location of containment failure, the timing of containment failure, and the occurrence of core-concrete interactions. Descriptions of the CAPB are presented in Table E.1-9.

Table E.1-9
Collapsed Accident Progression Bins (CAPB) Descriptions

CAPB Number	Description
CAPB-1	[CD, No VB, No CF, No CCI] Core damage (CD) occurs, but timely recovery of RPV injection prevents vessel breach (No VB). Therefore, containment integrity is not challenged (No CF) and core-concrete interactions are precluded (No CCI). However, the potential exists for in-vessel release to the environment due to containment design leakage.
CAPB-2	[CD, VB, No CF, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment does not fail structurally and is not vented (No CF). Ex-vessel releases are recovered, precluding core-concrete interactions (No CCI). Although containment does not fail, vessel breach does occur, therefore the potential exists for in- and ex-vessel releases to the environment due to containment design leakage. RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurs, containment does not fail.
CAPB-3	[CD, VB, No CF, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment does not fail structurally and is not vented (No CF). However, ex-vessel releases are not recovered in time, and therefore core-concrete interactions occur (CCI). RPV pressure is not important because, even though high pressure induced severe accident phenomena (such as direct containment heating [DCH]) occurs, containment does not fail, nor is the vent limit reached.
CAPB-4	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.

Table E.1-9
Collapsed Accident Progression Bins (CAPB) Descriptions
(Continued)

CAPB Number	Description
CAPB-5	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
CAPB-6	[CD, VB, Early CF, WW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). Following containment failure, core-concrete interactions occur (CCI).
CAPB-7	[CD, VB, Early CF, WW, RPV pressure <200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occur (CCI).
CAPB-8	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.

Table E.1-9
Collapsed Accident Progression Bins (CAPB) Descriptions
(Continued)

CAPB Number	Description
CAPB-9	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
CAPB-10	[CD, VB, Early CF, DW, RPV pressure >200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is greater than 200 psig at time of vessel breach (this implies that high pressure induced severe accident phenomena [DCH] are possible). Following containment failure, core-concrete interactions occur (CCI).
CAPB-11	[CD, VB, Early CF, DW, RPV pressure <200 psig at VB, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails either before core damage, during core damage, or at vessel breach (Early CF). Containment failure occurs in the drywell or below the torus water line (DW). RPV pressure is less than 200 psig at time of vessel breach; precluding high pressure induced severe accident phenomena. Following containment failure, core-concrete interactions occur (CCI).
CAPB-12	[CD, VB, Late CF, WW, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails late due to loss of containment heat removal (Late CF). Containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because high-pressure severe accident phenomena (such as DCH) did not fail containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.

Table E.1-9
Collapsed Accident Progression Bins (CAPB) Descriptions
(Continued)

CAPB Number	Description
CAPB-13	[CD, VB, Late CF, WW, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. Containment failure occurs in the torus (WW), above the water level. RPV pressure is not important because high-pressure severe accident phenomena (such as DCH) did not fail containment.
CAPB-14	[CD, VB, Late CF, DW, No CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails late due to loss of containment heat removal (Late CF). Containment failure occurs in the drywell or below the torus water level (DW). RPV pressure is not important because high-pressure severe accident phenomena did not fail containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
CAPB-15	[CD, VB, Late CF, DW, CCI] Core damage (CD) occurs followed by vessel breach (VB). Containment fails late (late CF) due to core-concrete interactions (CCI) after vessel breach. Containment failure occurs in the drywell or below the torus water level (DW). RPV pressure is not important because high-pressure severe accident phenomena did not fail containment.
CAPB-16	[CD, VB, BYPASS, RPV pressure >200 psig, No CCI] Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.
CAPB-17	[CD, VB, BYPASS, RPV pressure <200 psig, No CCI] Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. There are no core concrete interactions (No CCI) due to the presence of an overlying pool of water.

Table E.1-9
Collapsed Accident Progression Bins (CAPB) Descriptions
(Continued)

CAPB Number	Description
CAPB-18	[CD, VB, BYPASS, RPV pressure >200 psig, CCI] Small break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at high RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).
CAPB-19	[CD, VB, BYPASS, RPV pressure <200 psig, CCI] Large break interfacing system LOCA outside containment occurs. Core damage (CD) and subsequent vessel breach (VB) results at low RPV pressure with a bypassed containment. Following vessel breach, core-concrete interaction occurs (CCI).

Based on the above binning methodology, the salient Level 2 results are summarized in Tables E.1-10 and E.1-11 respectively. Table E.1-10 summarizes the results of the CET quantification. This table identifies the total annual release frequency for each Level 2 release category. Table E.1-11 provides the frequency, time, duration, energy, and elevation of release for each CAPB.

Table E.1-10
Summary of PNPS Containment Event Tree Quantification

Release Category (Timing/Magnitude)	Release Frequency (/RY)
Late Low	4.53E-06
Late Medium	1.56E-06
Late High	0.00E-00
Early Low	3.32E-08
Early Medium	6.48E-08
Early High	1.13E-07
No Containment Failure	1.11E-07

Nomenclature

Timing

L (Late) - Greater than 24 hours
E (Early) - Less than 24 hours

Magnitude

NCF (Little to no release) - Less than 0.001% CsI
LO (Low) - 0.001 to 1% CsI
MED (Medium) - 1 to 10% CsI
HI (High) - Greater than 10% CsI

Table E.1-11
Collapsed Accident Progression Bin (CAPB) Source Terms

	CAPB	CAPB Frequency (/year)	Warning Time (sec)	Elevation (m)	Release Start (sec)	Release Duration (sec)	Release Energy (W)
1	CAPB-1	9.51E-08	3.98E+03	3.00E+01	2.20E+04	9.00E+03	2.61E+05
2	CAPB-2	1.27E-08	3.96E+03	3.00E+01	2.20E+04	9.00E+03	2.50E+05
3	CAPB-3	2.39E-09	3.96E+03	3.00E+01	2.20E+04	9.00E+03	2.50E+05
4	CAPB-4	3.29E-09	7.96E+03	3.00E+01	1.83E+04	3.56E+03	1.10E+07
5	CAPB-5	2.73E-09	1.31E+04	3.00E+01	2.53E+04	7.93E+03	8.34E+06
6	CAPB-6	7.95E-09	1.33E+04	3.00E+01	2.56E+04	8.11E+03	8.23E+06
7	CAPB-7	7.93E-09	1.38E+04	3.00E+01	2.61E+04	8.46E+03	8.03E+06
8	CAPB-8	2.06E-08	9.18E+03	3.00E+01	2.00E+04	4.59E+03	1.04E+07
9	CAPB-9	9.25E-09	9.21E+03	3.00E+01	2.44E+04	8.87E+03	4.18E+06
10	CAPB-10	8.53E-08	1.37E+04	3.00E+01	2.60E+04	8.40E+03	8.06E+06
11	CAPB-11	4.35E-08	1.37E+04	3.00E+01	2.60E+04	8.40E+03	8.06E+06
12	CAPB-12	1.70E-06	2.84E+04	3.00E+01	4.64E+04	9.00E+03	7.59E+06
13	CAPB-13	2.30E-09	9.14E+03	3.00E+01	2.71E+04	9.00E+03	1.80E+06
14	CAPB-14	2.26E-06	2.66E+04	3.00E+01	4.46E+04	9.00E+03	7.08E+06
15	CAPB-15	2.12E-06	2.81E+04	3.00E+01	4.62E+04	9.00E+03	7.60E+06
16	CAPB-16	1.18E-09	3.96E+03	3.00E+01	2.12E+04	9.00E+03	2.50E+05
17	CAPB-17	6.91E-09	3.96E+03	3.00E+01	2.14E+04	9.00E+03	2.50E+05
18	CAPB-18	4.61E-10	3.96E+03	3.00E+01	2.12E+04	9.00E+03	2.50E+05
19	CAPB-19	2.43E-08	3.96E+03	3.00E+01	2.18E+04	9.00E+03	2.50E+05

Table E.1-11
Collapsed Accident Progression Bin (CAPB) Source Terms (continued)

	Release Fractions								
	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1	1.99E-07	1.85E-07	1.85E-07	0.00E+00	1.24E-09	8.00E-09	5.01E-11	8.43E-11	1.70E-08
2	9.97E-05	4.81E-05	4.66E-05	1.76E-07	3.97E-07	4.00E-06	1.65E-08	5.15E-08	4.87E-06
3	9.97E-05	5.37E-05	4.97E-05	1.76E-06	5.80E-07	4.00E-06	2.37E-08	1.57E-07	4.95E-06
4	1.00E+00	4.90E-02	2.62E-02	4.18E-05	2.46E-05	3.66E-04	8.97E-07	3.04E-06	1.92E-04
5	9.85E-01	7.86E-02	3.68E-02	4.28E-05	4.10E-05	3.66E-04	1.56E-06	6.79E-06	3.44E-04
6	1.00E+00	4.02E-02	2.32E-02	1.48E-03	3.19E-04	3.66E-04	6.50E-06	7.17E-05	3.23E-04
7	9.76E-01	6.11E-02	2.94E-02	1.26E-03	2.30E-04	3.66E-04	9.10E-06	1.06E-04	4.52E-04
8	1.00E+00	2.98E-01	2.72E-01	3.07E-05	9.89E-04	2.23E-02	4.49E-05	6.57E-05	1.15E-02
9	5.97E-01	7.61E-02	7.07E-02	1.41E-05	9.72E-04	1.09E-02	3.69E-05	7.63E-05	1.02E-02
10	1.00E+00	2.80E-01	2.49E-01	1.11E-02	3.07E-03	1.81E-02	7.95E-05	5.81E-04	1.03E-02
11	9.79E-01	1.73E-01	1.41E-01	9.97E-03	3.13E-03	1.78E-02	1.22E-04	9.39E-04	1.72E-02
12	2.01E-01	5.84E-05	4.37E-05	1.25E-07	2.36E-07	1.72E-06	8.04E-09	2.56E-08	2.99E-06
13	9.97E-01	7.99E-03	5.99E-03	1.76E-04	3.63E-05	3.66E-04	2.15E-06	1.41E-05	4.52E-04
14	7.75E-01	2.88E-02	2.67E-02	2.47E-05	2.05E-04	2.13E-03	8.49E-06	2.27E-05	2.61E-03
15	9.97E-01	2.76E-01	2.68E-01	1.27E-03	2.27E-03	2.25E-02	9.33E-05	3.00E-04	2.74E-02
16	1.00E+00	6.71E-02	3.26E-02	4.06E-04	9.11E-05	2.21E-02	1.45E-06	1.65E-05	4.27E-05
17	9.72E-01	3.62E-01	3.37E-01	1.34E-03	2.37E-03	2.20E-02	9.90E-05	1.62E-04	8.57E-03
18	1.00E+00	9.76E-02	6.25E-02	2.09E-02	4.67E-03	2.27E-02	7.45E-05	8.50E-04	2.12E-03
19	9.72E-01	4.03E-01	3.77E-01	6.87E-02	9.58E-03	2.26E-02	3.00E-04	2.33E-03	1.20E-02

E.1.2.2.7 Release Magnitude Calculations

The MAAP computer code is used to assign both the radionuclide release magnitude and timing based on the accident progression characterization. Specifically, MAAP provides the following information:

- containment pressure and temperature versus time (time of containment failure is determined by comparing these values with the nominal containment capability);
- radionuclide release time and magnitude for a large number of radioisotopes; and
- release fractions for twelve radionuclide species.

E.1.3 IPEEE Analysis

E.1.3.1 Seismic Analysis

PNPS performed a seismic PRA following the guidance of NUREG-1407, *Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities*, June 1991. The seismic PRA model was performed in conjunction with the SQUG program in 1994 as part of the IPEEE submittal report [Reference E.1-6]. The seismic, high wind, and external flooding analyses determined that the plant is adequately designed to protect against the effects of these natural events.

A number of plant improvements were identified in Table 2.4 of NUREG-1742, *Perspectives Gained from the IPEEE Program*, Final Report, April 2002 [Reference E.1-8]. These improvements were implemented.

The seismic CDF in the IPEEE was conservatively estimated to be 5.82×10^{-5} per reactor-year. The seismic CDF has recently been re-evaluated to reflect the updated Gothic computer code room heat up calculations that predict no room cooling requirements for HPCI, RCIC, Core Spray, and RHR areas; to update random component failure probabilities; and to model replacement of certain relays with a seismically rugged model. The updated seismic CDF of 3.22×10^{-5} per reactor-year was used in estimation of the factor of 6 used to determine the upper bound estimated benefit described in Section 4.21.5.4.

E.1.3.2 Fire Analysis

The PNPS internal fire risk model was performed in 1994 as part of the IPEEE submittal report [Reference E.1-6]. The PNPS fire analysis was performed using the conservative EPRI's Fire Induced Vulnerability Evaluation (FIVE) methodology for qualitative and quantitative screening of fire areas and for fire analysis of areas that did not screen [Reference E.1-7]. The FIVE methodology is primarily a screening approach used to identify plant vulnerabilities due to fire initiating events.

Table E.1-12 presents the results of the PNPS IPEEE fire analysis. The values presented in Table E.1-12 are taken from NUREG-1742 [Reference E.1-8]. These values are the same as the original IPEEE fire CDF results (2.20E-5 per reactor-year) [Reference E.1-6] after the response to NRC questions/issues regarding fire-modeling progression. A revised fire zone CDF of 1.91E-5 per reactor-year, generated to reflect updated equipment failure probability and unavailability values was used in estimation of the factor of 6 used to determine the upper bound estimated benefit described in Section 4.21.5.4.

The significant fire scenarios involve fires occurring in the train B switchgear room, turbine building heater bay, vital motor generator set room, and train A switchgear room.

Table E.1-12
PNPS Fire Updated Core Damage Frequency Results

Fire Compartment Sub-Area	Description	CDF/year	New Estimate CDF/year
1E	Reactor Building West, El. 21	9.7E-07	8.25E-07
2B	Turbine Building Heater Bay	2.1E-06	2.74E-06
3A	Train B RBCCW/TBCCW Pump and Heat Exchanger Room	2.0E-06	1.31E-06
4A	Train A RBCCW/TBCCW Pump and Heat Exchanger Room	9.8E-07	2.95E-07
6	Control Room	1.6E-06	8.90E-07
7	Cable Spreading Room	9.5E-07	7.85E-07
9	Vital Motor Generator Set Room	2.4E-06	2.38E-06
12	Train A Switchgear Room	3.1E-06	2.30E-06
13	Train B Switchgear Room	6.1E-06	6.85E-06
26	Main Transformer	1.5E-06	7.60E-07
		2.2E-05	**1.91E-05**

E.1.3.3 Other External Hazards

The PNPS IPEEE submittal [Reference E.1-6], in addition to the internal fires and seismic events, examined a number of other external hazards:

- high winds and tornadoes;
- external flooding; and
- ice, hazardous chemical, transportation, and nearby facility incidents.

In consequence of the above external hazards evaluation, no plant modifications were required for PNPS.

No risks to the plant occasioned by high winds and tornadoes, external floods, ice, and hazardous chemical, transportation, and nearby facility incidents were identified that might lead to core damage with a predicted frequency in excess of 10^{-6}/year. Therefore, these other external event hazards are not included in this attachment and are expected not to impact the conclusions of this SAMA evaluation.

E.1.4 PSA Model Peer Review and Difference between Current PSA Model and 1995 Update IPE

E.1.4.1 PSA Model Peer Review

The original IPE PSA model was peer reviewed on March 2000 using the BWROG PSA Peer Review Certification Implementation Guidelines. Facts and Observation sheets documented the certification teams' insights and potential level of significance. As part of the update of the IPE PSA models, all major issues and observations from the BWROG Peer Review (i.e., Level A, B, C, and D observations) have been addressed and incorporated into the current IPE PSA model, April 2003 [Reference E.1-1].

For the current IPE/PSA model update, individual work packages (event tree, fault tree, human reliability analysis (HRA), data, etc.) and internal flooding analysis were circulated to each PSA member for independent peer review. The accident sequence packages, system work packages, HRA, and internal flooding analyses were also assigned to the appropriate PNPS plant personnel for review. For example, event trees, system analyses, and fault tree models were forwarded to the applicable plant systems engineers and the HRA was assigned to individuals from the plant Operations Training department for review. Similarly, the accident sequence packages, system work packages, HRA report, containment performance analysis, fault tree and event tree models, and Level 2 models were peer reviewed by an outside consultant.

The Entergy license renewal project team and plant staff reviewed consequence and risk estimates for the SAMA analyses.

The peer review process emphasized the role of plant staff, external consultants, and BWROG PSA certification in this recent model update. The peer reviews served to ensure the accuracy of both the assumptions made in the models and the results. The results of the peer review and resolutions are presented in Section 5 and Appendix P of the Pilgrim Nuclear Power Station Individual Plant Examination for Internal Events update report, April 2003 [Reference E.1-1].

E.1.4.2 Major Differences between the Updated IPE PSA Model and 1995 Update IPE Model

E.1.4.2.1 Core Damage – Comparison to the PNPS 1995 Update IPE Model

The current PNPS IPE/PSA update model was completely revised in response to the BWROG Peer Review of March 2000 [Reference E.1-1]. The updated model is based upon all procedures and plant design as of September 30, 2001, and plant data as of December 31, 2001. The results yield a measurably lower CDF (point estimate CDF - 6.41E-6/reactor year) than the original IPE (point estimate CDF - 5.85E-5/yr) [Reference E.1-2] and 1995 PSA model update (point estimate CDF - 2.84E-5/yr) [Reference E.1-3]. (The 1995 update was performed to answer NRC questions following the IPE submittal.) The improved results are due to improved plant performance, replacement of switchyard breakers, more realistic success criteria based on MAAP runs, and more sophisticated data handling. Major changes are summarized as follows.

A. Initiating Event

The initiating event frequencies were updated to include current plant data and recent NRC publication information. For example, the LOOP frequency decreased significantly from the original IPE frequency of 0.475/yr to the current value of 0.067/yr [Reference E.1-1], which reflects the decreased occurrence of LOOP events since 1990 and replacement of switchyard breakers. In addition, fault tree models were developed to calculate support system initiating event frequencies.

B. Accident Sequence Evaluation

Event trees from the original IPE were completely revised. BWROG certification findings and observations were incorporated into the revised event trees. Major facts and observations include the following.

(1) LOOP Event Tree

The LOOP event was completely revised to account for failure modes of HPCI/RCIC beyond 8 hours of operation; RPV depressurization on HCTL; and transfer to the SBO tree to address such items as premature battery depletion and AC recovery at 30 minutes and beyond.

(2) SBO Event Tree

Current update reflects GE load shed calculations and use of plant SBO procedures for DC load shedding.

(3) Inadvertent Stuck Open Relief Valve (IORV) Event Tree

The IORV event tree was modified to include RPV depressurization with two SRVs given high-pressure injection failure.

(4) LOCAs Event Trees

The update considers both HPCI and RCIC for small break LOCAs.

Large and medium LOCAs and subsequent ATWS are modeled as core damage end states in the updated model. Small break LOCAs and ATWS are treated as similar to transient-induced ATWS.

The vapor suppression system is considered during large LOCAs events.

(5) ATWS Event Tree

The revised ATWS tree reflects the potential for MSIV closure on low RPV level.

The revised ATWS model takes into consideration "inhibit ADS" and MSIV bypass issues. In addition, HRA values take into consideration ATWS accident progressions for RPV and containment conditions predicted by MAAP.

(6) Loss-of-Containment Heat Removal Sequences

The revised event trees model the potential impact from containment venting on low-pressure system operation. For example, no credit is given for core spray and LPCI if containment venting is required. In addition, other containment related phenomena, such as high torus temperatures (HPCI) and high containment pressures (RCIC, SRVs) are reflected in the updated event trees.

The update model only considers the DTV path for containment venting.

(7) ISLOCA Event Tree

NSAC-154 [Reference E.1-10] and NUREG/CR-5124 [Reference E.1-11] were used to reassess the ISLOCA analysis.

Success criteria for low-pressure injection during an ISLOCA are consistent with those used for small LOCAs.

The revised ISLOCA event tree credits use of condensate or fire water for large ISLOCA events provided that LPCI or core spray operation had previously occurred to provide initial RPV reflood.

(8) Other Changes

The revised event trees credit use of feedwater when appropriate.

Control Rod Drive system flow into the RPV is credited for sequences that involve loss of containment heat removal and subsequent requirement to control containment pressure with direct torus containment venting.

Consistent success criteria were employed for RPV depressurization for transients, medium LOCAs, and small LOCAs.

The revised PNPS IPE models are based on the BWROG EPGs/SAGs Revision 4 of the BWROG EPGs [Reference E.1-1].

Core damage definition has been revised to be consistent with the EPRI PSA Applications Guide [Reference E.1-12]. That is, core damage occurs when peak clad temperature exceeds 2200°F.

HPCI and RCIC use is based on a 24-hour mission time.

C. Thermal - Hydraulic (T-H) Analysis

T-H analysis has been completely revised and improved to better support the success criteria. The MAAP4 computer code [Reference E.1-4] was used to update and address the many issues raised by the BWROG certification team, such as the following.

- A basis was provided for the timing and discharge pressure (flow) adequacy when using the fire water system for successful mitigation during transients and small LOCAs.
- Success criteria for SORV are same as for non-SORV cases (2 SRVs are required for successful RPV depressurization).
- Consistent success criteria are used for RPV depressurization for transients, medium LOCAs, and small LOCAs.
- Plant specific calculations were performed to identify the plant response for single or double recirculation pump trip failures.
- The appropriateness of the core damage definition used in the update was verified.

- In addition to the MAAP4 code, the GOTHIC code [Reference E.1-13] was used to predict various room heatup rates for the reactor building, turbine building, switchgear room, and battery room.

D. System Analysis

System fault tree models from the original IPE were completely revised to reflect the as-built plant configuration. MAAP analyses were clearly identified to support the success criteria of these Level 1 models. More detailed modeling for the logic interlock was included in the system models. A detailed fault tree for the RPS was developed based on NUREG/CR-5500 [Reference E.1-9], which decreased the failure-to-scram probability from 3.0E-5/yr to 5.8E-6/yr.

E. Data Analysis

Component failure data, both generic and plant-specific, were reviewed and updated with more recent experience (the performance of risk significant systems HPCI and RCIC has greatly improved since the original IPE). Plant-specific data were adjusted for industry experience using Bayesian updates. Maintenance unavailability values were updated based on maintenance rule records from the system engineers. More recent common cause failure data and approach NUREG/CR-5497 [Reference E.1-14] were factored into this update. In particular, a more detailed and refined common-cause failure methodology (Alpha model) has been applied in this update. In addition, more common-cause equipment failure groups such as fans, dampers, transformers, DC power panels, and circuit breakers have been included in the analysis.

F. HRA

A complete revision of the HRA was performed to identify, quantify, and document the pre-initiator and post-initiator human errors (including recoveries). The updated HRA was performed using NUREG/CR-1278 [Reference E.1-15], also referred to as THERP. Screening values were only used for low-significance human errors. In addition, a detailed analysis was performed to treat dependencies between post-initiator errors.

G. Dependency Analysis

A complete revision of the internal flooding analysis was developed to systematically address spatial dependencies.

Dependency between pre-initiator human errors (such as miscalibration of instruments) was modeled. In addition, dependencies between multiple post-accident operator actions appearing in the same accident sequence were evaluated.

Detailed component dependency tables were developed to address the support systems associated with the modeled systems and components.

H. Structural Response

The ISLOCA frequency was revised.

RPV overpressure and capability of the reactor building were included in the Level 2 assessment.

I. Quantification

The truncation value was lowered to 1.0E-11.

Human Error Probability (HEP) dependencies and recovery actions in the cutsets were evaluated.

ATWS contribution decreased due to lower probability of failure to scram based on NUREG/CR-5500 [Reference E.1-9].

The HRA was completely revised to address a comment from the PSA Certification [Reference E.1-16] that many of the HEPs were not realistic using the previous methodology. In many cases (e.g., failure to perform DTV), the previous HEPs were judged to be overly conservative.

J. Internal Flooding Analysis

The internal flooding analysis from the original IPE was completely revised to include a detailed, systematic examination of the flood source and progression for each of the analyzed flooding scenarios. In addition, the updated internal flooding analysis considers the effects of spray on equipment.

K. Uncertainty Analysis

An uncertainty analysis was performed for this update.

E.1.4.2.2 Containment Performance – Comparison to the Original PNPS IPE Model

Containment performance analysis models were completely revised from the original IPE. Propagation of Level 1 cutsets to the Level 2 CET was developed. A detailed LERF model was developed to ensure that LERF calculations are consistent with the PSA Applications Guide and NRC requirements for RG 1.174 [Reference E.1-17]. Other salient items incorporated are the following.

- CET fault models were revised to ensure that mitigating systems were not degraded in the Level 1 sequence.
- CET fault tree models allowed credit for AC power recovery post core damage. This ensures that the models do not allow SBO core damage sequences to benefit from AC supported equipment in Level 2 without explicit consideration of AC power recovery.

- Shell melt-through phenomena were considered where applicable.
- Operator responses to key actions were reassessed to incorporate the probability for success given the containment conditions and Emergency Operating Procedure directions.
- Direct torus venting was considered post core damage.
- PNPS-specific primary containment structural evaluation was included in the CET. This also included a structural evaluation of torus failure due to dynamic loading during ATWS scenarios, torus break below the water line, and bellows seal capability.
- A reactor building bypass fault tree model was developed to assess the impact on the Level 2 analysis.

E.1.5 The MACCS2 Model - Level 3 Analysis

E.1.5.1 Introduction

SAMA evaluation relies on Level 3 PRA results to measure the effects of potential plant modifications. A Level 3 PRA model using the MACCS2 [Reference E.1-18] was created for PNPS. This model, which requires detailed site-specific meteorological, population, and economic data, estimates the consequences in terms of population dose and offsite economic cost. Risks in terms of population dose risk (PDR) and offsite economic cost risk (OECR) were also estimated in this analysis. Risk is defined as the product of consequence and frequency of an accidental release.

This analysis considers a base case and two sensitivity cases to account for variations in data and assumptions for postulated internal events. The base case uses estimated time and speed for evacuation. Sensitivity case 1 is the base case with delayed evacuation. Sensitivity case 2 is the base case with lower evacuation speed.

PDR was estimated by summing over all releases the product of population dose and frequency for each accidental release. Similarly, OECR was estimated by summing over all releases the product of offsite economic cost and frequency for each accidental release. Offsite economic cost includes costs that could be incurred during the emergency response phase and costs that could be incurred through long-term protective actions.

E.1.5.2 Input

The following sections describe the site-specific input parameters used to obtain the off-site dose and economic impacts for cost-benefit analyses.

E.1.5.2.1 Projected Total Population by Spatial Element

The total population within a 50-mile radius of PNPS was estimated for the year 2032, the end of the proposed license renewal period, for each spatial element by combining total resident population projections with transient population data obtained from Massachusetts and Rhode Island. Table E.1-13 shows the estimated population distribution.

Table E.1-13
Estimated Population Distribution within a 50-mile Radius

Sector	0-10 Miles	10-20 Miles	20-30 Miles	30-40 Miles	40-50 Miles	50-Mile Total
N	0	0	0	0	80474	80474
NNE	3	0	0	0	0	3
NE	3	0	0	0	0	3
ENE	3	0	33121	0	0	33124
E	5	0	33121	23185	0	56311
ESE	23	0	49682	92740	0	142445
SE	950	9936	115925	23185	0	149996
SSE	13289	69555	82803	0	0	165647
S	23695	99364	132485	84383	43397	383324
SSW	23695	49762	23696	23185	21699	142037
SW	23695	71088	277374	349491	114546	836194
WSW	23695	71088	277374	349491	183037	904685
W	22818	71088	277374	388324	286370	1045974
WNW	16494	71088	118481	303450	390150	899663
NW	11269	71088	195075	1529212	405561	2212205
NNW	5599	35544	43350	31295	321894	437682
Total	165236	619601	1659861	3197941	1847128	7489767

The 2000 U.S. Census Bureau data, together with Massachusetts and Rhode Island population projection data, was used to project county-level resident populations to the year 2032. Seasonal peak transient population was conservatively used to establish a transient/resident population ratio for each county within the 50-mile radius. The ratio was found to be decreasing over time. For purposes of this study, the total county level population values were estimated by

summing the year 2000 peak transient population of each county and the projected year 2032 permanent population of that county to obtain the 2032 total county population.

E.1.5.2.2 Land Fraction

The land fraction for each spatial element was estimated from the PNPS Emergency Planning Zone maps for radii of 2, 5, and 50 miles [Reference E.1-20].

E.1.5.2.3 Watershed Class

There are two watershed types in the 50-mile zone surrounding PNPS: ocean and land (watersheds) drained by rivers. There are no major lakes. The watershed index assigns "1" to any spatial element having a non-zero land fraction and "2" to all elements over the Atlantic Ocean or its bays.

E.1.5.2.4 Regional Economic Data

Region Index

Each spatial element was assigned to an economic region, defined in this report as a county. Where a spatial element covers portions of more than one county, it was assigned to that county having the most area within the element.

Regional Economic Data

County level economic data were obtained from the U.S. Department of Agriculture. The Census of Agriculture is conducted every five years and data from 1997 and 1992 were used to project the farm-related economic data for 2002.

VALWF - Value of Farm Wealth

MACCS2 requires an average value of farm wealth (dollars/hectare) for the 50-mile radius area around PNPS. The county-level farmland property value was used as a basis for deriving this value. VALWF is $23,578/hectare.

VALWNF- Value of Non-Farm Wealth

MACCS2 also requires an average value of non-farm wealth. The county-level non-farm property value was used as a basis for deriving this value. VALWNF is $189,041/ person.

Other economic parameters and their values are shown below. The values were obtained by adjusting the economic data from a past census given as default values in Reference E.1-18 with the consumer price index of 177.1, which is the average value for the year 2001, as appropriate.

Variable	Description	Value
EVACST	Daily cost for a person who has been evacuated ($/person-day)	42.3
POPCST	Population relocation cost ($/person)	7840
RELCST	Daily cost for a person who is relocated ($/person-day)	42.3
CDFRM0	Cost of farm decontamination for the various levels of decontamination ($/hectare)	881 1959
CDNFRM	Cost of non-farm decontamination for the various levels of decontamination ($/person)	4700 12540
DLBCST	Average cost of decontamination labor ($/person-year)	54800
DPRATE	Property depreciation rate (per year)	0.2
DSRATE	Investment rate of return (per year)	0.12

E.1.5.2.5 Agriculture Data

The source of regional crop information is the New England Agricultural Statistics, 2001. The crops listed for each of the two states, Massachusetts and Rhode Island, were mapped into the seven MACCS2 crop categories.

E.1.5.2.6 Meteorological Data

The MACCS2 model requires meteorological data for wind speed, wind direction, atmospheric stability, accumulated precipitation, and atmospheric mixing heights. The required data was obtained from the PNPS site meteorological monitoring system and the Automated Surface Observatory System (ASOS) at Plymouth Airport.

Site Specific Data

Site specific meteorological data is available from two meteorological towers, one located off the main parking lot and the second located west of the old I&S building, the "lower" and "upper" towers respectively. The upper tower is the designated data source for MACCS2 input. Data from the lower tower was used only if measurements from the upper tower were missing for a specific hour.

Year 2001 hourly data from the upper tower was used in this analysis. The data was more than 98% complete. Missing data was obtained either from the lower tower or from estimates based on adjacent valid measurements of the missing hour.

Accumulated Precipitation

The nearest source of hourly precipitation data to PNPS is the ASOS at Plymouth Airport. The data was converted to MACCS2 input format to provide precipitation in hundredths of an inch.

Regional Mixing Height Data

Mixing height is defined as the height of the atmosphere above ground level within which a released contaminant will become mixed (from turbulence) within approximately one hour. PNPS mixing height data, given in Reference E.1-19, was used for MACCS2 analysis.

E.1.5.2.7 Emergency Response Assumptions

Details of the evacuation time estimates including supporting assumptions regarding population, alarm criteria, delay times, areas, speed, distance, and routes are contained in the PNPS Emergency Plan [Reference E.1-20].

Evacuation Delay Time

The elapsed time between siren alert and the beginning of evacuation is 40 minutes. A sensitivity case that assumes 2 hours for evacuees to begin evacuation was considered in this study to evaluate consequence sensitivities due to uncertainties in delay time.

Evacuation Speed

The worst case for PNPS evacuation is during the winter, under adverse weather conditions, since snow removal can add up to an hour and a half to the evacuation time. The radius of the Emergency Planning Zone is 10 miles. Assuming that the net movement of the entire population is 10 miles, the time required for evacuation ranges from 3 hours 35 minutes to 6 hours 30 minutes, and the average evacuation speed ranges from 2.79 miles/hour in clear weather to 1.54 miles/hour under adverse weather conditions. The average evacuation speed is 2.17 miles/hour, or 0.97 meter/second.

A sensitivity case that assumes a lower evacuation speed of 0.69 meter/second was considered in this study to evaluate consequence sensitivities due to uncertainties in evacuation speed.

E.1.5.2.8 Core Inventory

The estimated PNPS core inventory (Table E.1-14) used in the MACCS2 input is based on a power level of 2028 MW(t).

Table E.1-14
PNPS Core Inventory (Becquerels)

Nuclide	Inventory	Nuclide	Inventory
Co-58	1.15E+16	Te-131m	2.87E+17
Co-60	1.37E+16	Te-132	2.80E+18
Kr-85	1.88E+16	I-131	1.94E+18
Kr-85m	6.84E+17	I-132	2.85E+18
Kr-87	1.24E+18	I-133	4.07E+18
Kr-88	1.68E+18	I-134	4.45E+18
Rb-86	1.05E+15	I-135	3.83E+18
Sr-89	2.08E+18	Xe-133	4.07E+18
Sr-90	1.47E+17	Xe-135	9.68E+17
Sr-91	2.71E+18	Cs-134	3.17E+17
Sr-92	2.83E+18	Cs-136	8.51E+16
Y-90	1.58E+17	Cs-137	1.90E+17
Y-91	2.54E+18	Ba-139	3.75E+18
Y-92	2.84E+18	Ba-140	3.70E+18
Y-93	3.23E+18	La-140	3.77E+18
Zr-95	3.34E+18	La-141	3.48E+18
Zr-97	3.44E+18	La-142	3.35E+18
Nb-95	3.16E+18	Ce-141	3.36E+18
Mo-99	3.65E+18	Ce-143	3.27E+18
Tc-99m	3.15E+18	Ce-144	2.18E+18
Ru-103	2.77E+18	Pr-143	3.20E+18
Ru-105	1.85E+18	Nd-147	1.43E+18
Ru-106	7.52E+17	Np-239	4.26E+19
Rh-105	1.38E+18	Pu-238	2.96E+15
Sb-127	1.74E+17	Pu-239	7.51E+14
Sb-129	6.06E+17	Pu-240	9.41E+14
Te-127	1.69E+17	Pu-241	1.62E+17
Te-127m	2.27E+16	Am-241	1.65E+14
Te-129	5.68E+17	Cm-242	4.35E+16
Te-129m	1.49E+17	Cm-244	2.35E+15

Source: derived from Reference E.1-21 for a power level of 2028 MW(t)

E.1.5.2.9 Source Terms

Twelve release categories, corresponding to internal event sequences, were part of the MACCS2 input. Details of the source terms for postulated internal events are available in on-site documentation. A linear release rate was assumed between the time the release started and the time the release ended.

E.1.5.3 Results

Risk estimates for one base case and two sensitivity cases were analyzed with MACCS2. The base case assumes 40 minute delay and 0.97 meter/sec speed of evacuation. Sensitivity case 1 is the base case with delayed evacuation of 2 hours. Sensitivity case 2 is the base case with an evacuation speed of 0.69 meter/sec.

Table E.1-15 shows estimated base case mean risk values for each release mode. The estimated mean values of PDR and offsite OECR for PNPS are 13.6 person-rem/yr and $45,900/yr, respectively.

Table E.1-15
Base Case Mean PDR and OECR Values

Release Mode	Frequency (/yr)	Population Dose (person-sv)[1]	Offsite Economic Cost ($)	Population Dose Risk (PDR) (person-rem/yr)	Offsite Economic Cost Risk (OECR) ($/yr)
CAPB-1	9.51E-08	4.66E-01	3.82E+06	4.43E-06[2]	3.63E-01
CAPB-2	1.27E-08	9.96E+01	6.40E+06	1.26E-04	8.10E-02
CAPB-3	2.39E-09	1.06E+02	6.48E+06	2.53E-05	1.55E-02
CAPB-4	3.29E-09	1.38E+04	4.28E+09	4.54E-03	1.41E+01
CAPB-5	2.73E-09	1.81E+04	5.30E+09	4.94E-03	1.45E+01
CAPB-6	7.95E-09	1.51E+04	3.51E+09	1.20E-02	2.79E+01
CAPB-7	7.93E-09	1.67E+04	4.42E+09	1.32E-02	3.51E+01
CAPB-8	2.06E-08	4.10E+04	1.47E+10	8.44E-02	3.03E+02
CAPB-9	9.25E-09	2.37E+04	8.33E+09	2.19E-02	7.70E+01
CAPB-10	8.53E-08	4.31E+04	1.54E+10	3.68E-01	1.31E+03
CAPB-11	4.35E-08	3.45E+04	1.15E+10	1.50E-01	5.00E+02
CAPB-12	1.70E-06	9.72E+01	4.63E+06	1.65E-02	7.88E+00
CAPB-13	2.30E-09	7.30E+03	6.53E+08	1.68E-03	1.50E+00
CAPB-14	2.26E-06	1.58E+04	4.14E+09	3.57E+00	9.36E+03
CAPB-15	2.12E-06	4.31E+04	1.59E+10	9.14E+00	3.37E+04
CAPB-16	1.18E-09	1.86E+04	5.50E+09	2.19E-03	6.48E+00
CAPB-17	6.91E-09	4.81E+04	1.71E+10	3.32E-02	1.18E+02
CAPB-18	4.61E-10	2.38E+04	7.86E+09	1.10E-03	3.62E+00
CAPB-19	2.43E-08	5.31E+04	1.88E+10	1.29E-01	4.56E+02
			Totals	**1.36E+01**	**4.59E+04**

1. 1 sv = 100 rem
2. 4.43E-06 (person-rem/yr) = 9.51E-08 (/yr) x 4.66E-01 (person-sv) x 100 (rem/sv)

Results of sensitivity analyses indicate that a delayed evacuation or a lower evacuation speed would not have significant effects on the offsite consequences or risks determined in this study. Table E.1-16 summarizes offsite consequences in terms of population dose (person-sv) and offsite economic cost ($) for the base case and the sensitivity cases. Comparison of the consequences indicates that the maximal deviation is less than 2% between the base case population dose and the Sensitivity Case 2 population dose for release mode CAPB-8.

Table E.1-16
Summary of Offsite Consequence Sensitivity Results

	Population Dose (person-sv)			Offsite Economic Cost ($)		
Release Mode	Base Case	2-Hr Delayed Evacuation	Lower Speed of Evacuation	Base Case	2-Hr Delayed Evacuation	Lower Speed of Evacuation
CAPB-1	4.66E-01	4.66E-01	4.67E-01	3.82E+06	3.82E+06	3.82E+06
CAPB-2	9.96E+01	9.97E+01	9.97E+01	6.40E+06	6.40E+06	6.40E+06
CAPB-3	1.06E+02	1.06E+02	1.06E+02	6.48E+06	6.48E+06	6.48E+06
CAPB-4	1.38E+04	1.39E+04	1.39E+04	4.28E+09	4.28E+09	4.28E+09
CAPB-5	1.81E+04	1.82E+04	1.82E+04	5.30E+09	5.30E+09	5.30E+09
CAPB-6	1.51E+04	1.51E+04	1.51E+04	3.51E+09	3.51E+09	3.51E+09
CAPB-7	1.67E+04	1.68E+04	1.68E+04	4.42E+09	4.42E+09	4.42E+09
CAPB-8	4.10E+04	4.16E+04	4.17E+04	1.47E+10	1.47E+10	1.47E+10
CAPB-9	2.37E+04	2.38E+04	2.39E+04	8.33E+09	8.33E+09	8.33E+09
CAPB-10	4.31E+04	4.34E+04	4.36E+04	1.54E+10	1.54E+10	1.54E+10
CAPB-11	3.45E+04	3.48E+04	3.49E+04	1.15E+10	1.15E+10	1.15E+10
CAPB-12	9.72E+01	9.75E+01	9.78E+01	4.63E+06	4.63E+06	4.63E+06
CAPB-13	7.30E+03	7.30E+03	7.31E+03	6.53E+08	6.53E+08	6.53E+08
CAPB-14	1.58E+04	1.58E+04	1.59E+04	4.14E+09	4.14E+09	4.14E+09
CAPB-15	4.31E+04	4.33E+04	4.35E+04	1.59E+10	1.59E+10	1.59E+10
CAPB-16	1.86E+04	1.87E+04	1.88E+04	5.50E+09	5.50E+09	5.50E+09
CAPB-17	4.81E+04	4.83E+04	4.86E+04	1.71E+10	1.71E+10	1.71E+10
CAPB-18	2.38E+04	2.39E+04	2.40E+04	7.86E+09	7.86E+09	7.86E+09
CAPB-19	5.31E+04	5.33E+04	5.37E+04	1.88E+10	1.88E+10	1.88E+10

E.1.6 References

E.1-1 ENN Engineering Report PNPS-PSA, "Pilgrim Nuclear Power Station Individual Plant Examination for Internal Events Update," April 2003, Revision 1.

E.1-2 Pilgrim Nuclear Power Station Individual Plant Examination, Revision 0, September 1992.

E.1-3 Boston Edison Company to the NRC, Response to Request for Additional Information Regarding the Pilgrim Individual Plant Examination (IPE) Submittal (TAC No. M74451, letter dated December 28, 1995 (2.95.127).

E.1-4 Modular Accident Analysis Program Boiling Water Reactor (MAAP BWR) Code, Version 4.0.4 and Fauske & Associates, Inc., "MAAP 4.0 Users manual," prepared for The Electric Power Research Institute, May 1994.

E.1-5 Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," Executive Conference on the Ramifications of the Source Term (Charleston, SC), March 12, 1985.

E.1-6 "Pilgrim Nuclear Power Station Individual Plant Examination for External Events," July 1994, Revision 0.

E.1-7 Parkinson, W. J., "EPRI Fire PRA Implementation Guide", prepared by Science Applications International Corporation for Electric Power Research Institute, EPRI TR-105928, December 1995.

E.1-8 U.S. Nuclear Regulatory Commission, NUREG-1742, *Perspectives Gained From the Individual Plant Examination of External Events (IPEEE) Program*, Volume 1, Final Report, April 2002.

E.1-9 U.S. Nuclear Regulatory Commission, NUREG/CR-5500, Vol. 3, (INEEL/EXT-97-00740), *Reliability Study: General Electric Reactor Protection System, 1984-1995*, May 1999.

E.1-10 Electric Power Research Institute, NSAC-154, "ISLOCA Evaluation Guidelines," prepared by ERIN Engineering and Research, Inc., September 1991.

E.1-11 Chu, et al., "Interfacing Systems LOCA: Boiling Water Reactors," Brookhaven National Laboratory, NUREG/CR-5124, BNL-NUREG-52141, February 1989.

E.1-12 Electric Power Research Institute, "PSA Applications Guide," EPRI TR-105396, prepared by ERIN Engineering and Research, Inc., August, 1995.

E.1-13 GOTHIC Containment Analysis Package, Version 3.4e, EPRI Tr-103053-V2, October 1993.

E.1-14 U.S. Nuclear Regulatory Commission, NUREG/CR-5497, (INEEL/EXT-97-01328), *Common-Cause Failure Parameter Estimations*, October 1998.

E.1-15 Swain, A. D. and H. E. Guttmann, NUREG/CR-1278, *Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications*, Sandia National Laboratories, U.S. Nuclear Regulatory Commission, August 1983.

E.1-16 BWR Owners Group, "Pilgrim PSA Certification," BWROG/PSA-9903, March 2000.

E.1-17 U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174 (draft was issued as DG-1061), "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July 1998.

E.1-18 Chanin, D. I., and M. L. Young, Code Manual for MACCS2: Volume 1, User's Guide, SAND97-0594 Sandia National Laboratories, Albuquerque, NM, 1997.

E.1-19 Boston Edison Company, "Appendix I Evaluation," forwarding evaluation of Pilgrim Station Unit 1 Conformance to the Design Objectives of 10 CFR 50, Appendix I, letter dated March 31, 1977 (2.77.031).

E.1-20 PNPS Emergency Plan, Revision 24, February 7, 2001, Appendix 5, Pilgrim Station Evacuation Time Estimates and Traffic Management Plan Update, Revision 5, November 1998.

E.1-21 U.S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 2, Rev. 1, Part 7, *Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, MACCS Input*, December 1990.

Exhibit No. NRC000001
Pilgrim LR Proceeding
50-293-LR, 06-848-02-LR

ATTACHMENT E.2

SAMA CANDIDATES SCREENING AND EVALUATION

E.2 EVALUATION OF SAMA CANDIDATES

This section describes the generation of the initial list of potential SAMA candidates, screening methods, and the analysis of the remaining SAMA candidates.

E.2.1 SAMA List Compilation

A list of SAMA candidates was developed by reviewing industry documents and considering plant-specific enhancements not identified in published industry documents. Since PNPS is a conventional GE nuclear power reactor design, considerable attention was paid to the SAMA candidates from SAMA analyses for other GE plants. Industry documents reviewed include the following:

- Hatch SAMA Analysis (Reference E.2-1),
- Calvert Cliffs Nuclear Power Plant SAMA Analysis (Reference E.2-2),
- GE ABWR SAMDA Analysis (Reference E.2-3),
- Peach Bottom SAMA Analysis (Reference E.2-4),
- Quad Cities SAMA Analysis (Reference E.2-5),
- Dresden SAMA Analysis (Reference E.2-6), and
- Arkansas Nuclear Unit 2 SAMA Evaluation (Reference E.2-7).

The above documents represent a compilation of most SAMA candidates developed from the industry documents. These sources of other industry documents include the following:

- Limerick SAMDA cost estimate report (Reference E.2-8),
- NUREG-1437 description of Limerick SAMDA (Reference E.2-9),
- NUREG-1437 description of Comanche Peak SAMDA (Reference E.2-10),
- Watts Bar SAMDA submittal (Reference E.2-11),
- TVA's response to NRC's RAI on the Watts Bar SAMDA submittal (Reference E.2-12),
- Westinghouse AP600 SAMDA (Reference E.2-13),
- NUREG-0498, Watts Bar Final Environmental Statement Supplement 1, Section 7 (Reference E.2-14),
- NUREG-1560, Volume 2, NRC Perspectives on the IPE Program (Reference E.2-15), and
- NUREG/CR-5474, Assessment of Candidate Accident Management Strategies (Reference E.2-16).

In addition to SAMA candidates from review of industry documents, additional SAMA candidates were obtained from plant-specific sources, such as the PNPS IPE (Reference E.2-17) and IPEEE (Reference E.2-18). In both the IPE and IPEEE, several enhancements related to severe accident insights were recommended and implemented. These enhancements are included in the comprehensive list of phase I SAMA candidates as numbers 248 through 281. The current PNPS PSA model was also used to identify plant-specific modifications for inclusion in the comprehensive list of SAMA candidates. The risk-significant terms from the current PSA model were reviewed for similar failure modes and effects that could be addressed through a potential enhancement to the plant. The correlation between SAMAs and the risk-significant terms were listed in Table E.1-2.

The comprehensive list, available in on-site documentation, contained a total of 281 phase I SAMA candidates.

E.2.2 Qualitative Screening of SAMA Candidates (Phase I)

The purpose of the preliminary SAMA screening was to eliminate from further consideration enhancements that were not viable for implementation at PNPS. Potential SAMA candidates were screened out if they modified features not applicable to PNPS, if they had already been implemented at PNPS, or if they were similar in nature and could be combined with another SAMA candidate to develop a more comprehensive or plant-specific SAMA candidate. During this process, 63 of the phase I SAMA candidates were screened out because they were not applicable to PNPS, 4 of the phase I SAMA candidates were screened out because they were similar in nature and could be combined with another SAMA candidate, and 155 of the phase I SAMA candidates were screened out because they had already been implemented at PNPS, leaving 59 SAMA candidates for further analysis. The final screening process involved identifying and eliminating those items whose implementation cost would exceed their benefit as described below. Table E.2-1 provides a description of each of the 59 phase II SAMA candidates.

E.2.3 Final Screening and Cost Benefit Evaluation of SAMA Candidates (Phase II)

A cost/benefit analysis was performed on each of the remaining SAMA candidates. If the implementation cost of a SAMA candidate was determined to be greater than the potential benefit (i.e. there was a negative net value) the SAMA candidate was considered not to be cost beneficial and was not retained as a potential enhancement.

The expected cost of implementation of each SAMA was established from existing estimates of similar modifications. Most of the cost estimates were developed from similar modifications considered in previously performed SAMA and SAMDA analyses. In particular, these cost-estimates were derived from the following major sources:

- GE ABWR SAMDA Analysis (Reference E.2-3),
- Peach Bottom SAMA Analysis (Reference E.2-4),

- Quad Cities SAMA Analysis (Reference E.2-5),
- Dresden SAMA Analysis (Reference E.2-6),
- ANO-2 SAMA Analysis (Reference E.2-7), and

The cost estimates did not include the cost of replacement power during extended outages required to implement the modifications, nor did they include contingency costs associated with unforeseen implementation obstacles. Estimates based on modifications that were implemented or estimated in the past were presented in terms of dollar values at the time of implementation (or estimation), and were not adjusted to present-day dollars. In addition, several implementation costs were originally developed for SAMDA analyses (i.e., during the design phase of the plant), and therefore, do not capture the additional costs associated with performing design modifications to existing plants (i.e., reduced efficiency, minimizing dose, disposal of contaminated material, etc.). Therefore, the cost estimates were conservative.

The benefit of implementing a SAMA candidate was estimated in terms of averted consequences. The benefit was estimated by calculating the arithmetic difference between the total estimated costs associated with the four impact areas for the baseline plant design and the total estimated impact area costs for the enhanced plant design (following implementation of the SAMA candidate).

Values for avoided public and occupational health risk were converted to a monetary equivalent (dollars) via application of the NUREG/BR-0184 (Reference E.2-19) conversion factor of $2,000 per person rem and discounted to present value. Values for avoided off-site economic costs were also discounted to present value.

As this analysis focuses on establishing the economic viability of potential plant enhancement when compared to attainable benefit, detailed cost estimates often were not required to make informed decisions regarding the economic viability of a particular modification. Several of the SAMA candidates were clearly in excess of the attainable benefit estimated from a particular analysis case.

For less clear cases, engineering judgment on the cost associated with procedural changes, engineering analysis, testing, training, and hardware modification was applied to determine if a more detailed cost estimate was necessary to formulate a conclusion regarding the economic viability of a particular SAMA. Based on a review of previous submittals' SAMA evaluations and an evaluation of expected implementation costs at PNPS, the following estimated costs for each potential element of the proposed SAMA implementation are used.

Type of Change	Estimated Cost Range
Procedural only	$25K-$50K
Procedural change with engineering required	$50K-$200K
Procedural change with engineering and testing/training required	$200K-$300K
Hardware modification	$100K to >$1000K

In most cases, more detailed cost estimates were not required, particularly if the SAMA called for the implementation of a hardware modification. Nonetheless, the cost of each unscreened SAMA candidate was conceptually estimated to the point where conclusions regarding the economic viability of the proposed modification could be adequately gauged. The cost benefit comparison and disposition of each of the 59 phase II SAMA candidates is presented in Table E.2-1.

Bounding evaluations (or analysis cases) were performed to address specific SAMA candidates or groups of similar SAMA candidates. These analysis cases overestimated the benefit and thus were conservative calculations. For example, one SAMA candidate suggested installing a digital large break LOCA protection system. The bounding calculation estimated the benefit of this improvement by total elimination of risk due to large break LOCA (see analysis in phase II SAMA 052 of Table E.2-1). This calculation obviously overestimated the benefit, but if the inflated benefit indicated that the SAMA candidate was not cost beneficial, then the purpose of the analysis was satisfied.

A description of the analysis cases used in the evaluation follows.

Decay Heat Removal Capability - Torus Cooling

This analysis case was used to evaluate the change in plant risk from installing an additional decay heat removal system. Enhancements of decay heat removal capability decrease the probability of loss of containment heat removal. A bounding analysis was performed by setting the events for loss of the torus cooling mode of the RHR system to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $261,832. This analysis case was used to model the benefit of phase II SAMAs 1 and 14.

Decay Heat Removal Capability - Drywell Spray

This analysis case was used to evaluate the change in plant risk from installing an additional decay heat removal system. Enhancements of decay heat removal capability decrease the

probability of loss of containment heat removal. A bounding analysis was performed by setting the events for loss of the drywell spray mode of the RHR system to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $264,219. This analysis case was used to model the benefit of phase II SAMA 9.

Filtered Vent

This analysis case was used to evaluate the change in plant risk from installing a filtered containment vent to provide fission product scrubbing. A bounding analysis was performed by reducing the successful torus venting accident progression source terms by a factor of 2 to reflect the additional filtered capability. Reducing the releases from the vent path resulted in no benefit. This analysis case was used to model the benefit of phase II SAMAs 2 and 19.

Containment Vent for ATWS Decay Heat Removal

This analysis case was used to evaluate the change in plant risk from installing a containment vent to provide alternate decay heat removal capability during an ATWS event. A bounding analysis was performed by setting the ATWS sequences associated with containment bypass to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $61,701. This analysis case was used to model the benefit of phase II SAMAs 3 and 47.

Molten Core Debris Removal

This analysis case was used to estimate the change in plant risk from providing a molten core debris cooling mechanism. A bounding analysis was performed by setting containment failure due to core-concrete interaction (not including liner failure) to zero in the level 2 PSA model, which resulted in an upper bound benefit of approximately $2,620,551. This analysis case was used to model the benefit of phase II SAMAs 4, 5, 8, and 23.

Drywell Head Flooding

This analysis case was used to evaluate the change in plant risk from providing a modification to flood the drywell head such that if high drywell temperature occurred, the drywell head seal would not fail. A bounding analysis was performed by setting the probability of drywell head failure due to high temperature to zero in the level 2 PSA model, which resulted in an upper bound benefit of approximately $12,915. This analysis case was used to model the benefit of phase II SAMAs 6, 18, and 20.

Reactor Building Effectiveness

This analysis case was used to evaluate the change in plant risk by ensuring the reactor building is available to provide effective fission product removal. Reactor building effectiveness was conservatively modeled by assuming reactor building availability for all accident sequences. This resulted in an upper bound benefit of approximately $64,577. This analysis case was used to model the benefit of phase II SAMAs 7, 13, and 21.

Strengthen Containment

This analysis case was used to evaluate the change in plant risk from strengthening containment to reduce the probability of containment over-pressurization failure. A bounding analysis was performed by setting all energetic containment failure modes (DCH, steam explosions, late over-pressurization) to zero in the level 2 PSA model, which resulted in an upper bound benefit of approximately $1,233,428. This analysis case was used to model the benefit of phase II SAMAs 10, 15, 16, and 24.

Base Mat Melt-Through

This analysis case was used to evaluate the change in plant risk from increasing the depth of the concrete base mat to ensure base mat melt-through does not occur. A bounding analysis was performed by setting containment failure due to base mat melt-through to zero in the level 2 PSA model, which resulted in an upper bound benefit of approximately $25,831. This analysis case was used to model the benefit of phase II SAMA 11.

Reactor Vessel Exterior Cooling

This analysis case was used to evaluate the change in plant risk from providing a method to perform ex-vessel cooling of the lower reactor vessel head. A bounding analysis was performed by modifying the probability of vessel failure by a factor of two to account for ex-vessel cooling in the level 2 PSA model, which resulted in an upper bound benefit of approximately $19,373. This analysis case was used to model the benefit of phase II SAMA 12.

Vacuum Breakers

This analysis case was used to evaluate the change in plant risk from improving the reliability of vacuum breakers to reseat following a successful opening and eliminate suppression pool scrubbing failures from the containment analysis. A bounding analysis was performed by setting the vacuum breaker failure probability to zero in the level 1 PSA model, which resulted in no benefit. This analysis case was used to model the benefit of phase II SAMA 17.

Flooding the Rubble Bed

This analysis case was used to evaluate the change in plant risk from providing a source of water to the drywell floor to flood core debris. A bounding analysis was performed by substituting the probabilities of wet core concrete interactions for dry core concrete interactions in the level 2 PSA model, which resulted in an upper bound benefit of approximately $1,226,971. This analysis case was used to model the benefit of phase II SAMA 22.

DC Power

This analysis case was used to evaluate the change in plant risk from plant modifications that would increase the availability of Class 1E DC power (e.g., increasing battery capacity, using fuel cells, or extending SBO injection provisions). It was assumed that battery life could be extended

from 14 hours to 24 hours to simulate additional battery capacity. This enhancement would extend HPCI and RCIC operability and allow more credit for AC power recovery. A bounding analysis was performed by changing the time available to recover offsite power before HPCI and RCIC are lost from 14 hours to 24 hours during SBO scenarios in the level 1 PSA model. This resulted in an upper bound benefit of approximately $146,356. This analysis case was used to model the benefit of phase II SAMAs 25, 26, 28, 33, and 35.

Improve DC System

This analysis case was used to evaluate the change in plant risk from improving injection capability by auto-transfer of AC bus control power to a standby DC power source upon loss of the normal DC source or from enhancing procedure to make use of DC bus cross-tie to improve DC power availability and reliability. A bounding analysis was performed by setting the DC buses D16 and D17 to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $118,568. This analysis case was used to model the benefit of phase II SAMAs 27 and 34.

Alternate Pump Power Source

This analysis case was used to evaluate the change in plant risk from adding 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. A bounding analysis was performed by setting failure of the SBO diesel generator to zero in level 1 PSA model, which resulted in an upper bound benefit of approximately $265,687. This analysis case was used to model the benefit of phase II SAMA 29.

Improve AC Power System

This analysis case was used to evaluate the change in plant risk from improving AC power system cross-tie capability to enhance the availability and reliability of the AC power system. A bounding analysis was performed by setting the loss of MCCs B17, B18, and B15 to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $473,410. This analysis case was used to model the benefit of phase II SAMA 30.

Dedicated DC Power and Additional Batteries and Divisions

This analysis case was used to evaluate the change in plant risk from plant modifications that would provide motive power to components (e.g., providing a dedicated DC power supply, additional batteries, or additional divisions). A bounding analysis was performed by setting the loss of DC bus D17 initiator, and one division of DC power, to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $903,025. This analysis case was used to model the benefit of phase II SAMAs 31 and 32.

Locate RHR Inside Containment

This analysis case was used to evaluate the change in plant risk from moving the RHR system inside containment to prevent an RHR system ISLOCA event outside containment. A bounding analysis was performed by setting the RHR ISLOCA sequences to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $16,497. This analysis case was used to model the benefit of phase II SAMA 36.

ISLOCA

This analysis case was used to evaluate the change in plant risk from reducing the probability of an ISLOCA by increasing the frequency of valve leak testing. A bounding analysis was performed by setting the ISLOCA initiator to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $24,148. This analysis case was used to model the benefit of phase II SAMA 37.

MSIV Design

This analysis case was used to evaluate the change in plant risk from improving MSIV design to decrease the likelihood of containment bypass scenarios. A bounding analysis was performed by setting the containment bypass failure due to MSIV leakage to zero in the level 2 PSA model, which resulted in no benefit. This analysis case was used to model the benefit of phase II SAMA 38.

Diesel to CST Makeup Pumps

This analysis case was used to evaluate the change in plant risk from installing an independent diesel for the CST makeup pumps to allow continued operation of the high pressure injection system during an SBO event. As currently modeled, if CST water level is low, swapping HPCI/ RCIC suction from the CST to the torus allows continued HPCI and RCIC injection. Therefore, a bounding analysis was performed by setting the failure to switchover from CST to torus to zero in the level 1 PSA model, which resulted in no benefit. This analysis case was used to model the benefit of phase II SAMA 39.

High Pressure Injection System

This analysis case was used to evaluate the change in plant risk from plant modifications that would increase the availability of high pressure injection (e.g., installing an independent AC powered high pressure injection system, passive high pressure injection system, or an additional high pressure injection system). A bounding analysis was performed by setting the CDF contribution due to unavailability of the HPCI system to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $110,212. This analysis case was used to model the benefit of phase II SAMAs 40, 41, 42, 44, and 45.

Improve the Reliability of High Pressure Injection System

This analysis case was used to evaluate the change in plant risk from plant modifications that would increase the reliability of the high pressure injection system. A bounding analysis was performed by reducing the HPCI system failure probability by a factor of three in the level 1 PSA model, which resulted in an upper bound benefit of approximately $76,025. This analysis case was used to model the benefit of phase II SAMA 43.

SRVs Reseat

This analysis case was used to evaluate the change in plant risk from improving the reliability of SRVs reseating. A bounding analysis was performed by setting the stuck open SRVs initiator to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $63,599. This analysis case was used to model the benefit of phase II SAMA 46.

Diversity of Explosive Valves

This analysis case was used to evaluate the change in plant risk from providing an alternate means of opening a pathway to the RPV for SLC system injection, thereby improving success probability for reactor shutdown. A bounding analysis was performed by setting common cause failure of SLC explosive valves to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $12,915. This analysis case was used to model the benefit of phase II SAMA 48.

Reliability of SRVs

This analysis case was used to evaluate the change in plant risk from installing additional signals to automatically open the SRVs. This improvement would reduce the likelihood of SRVs failing to open, thereby reducing the consequences of medium LOCAs. A bounding analysis was performed by setting the probability of SRVs failing to open when required by reactor pressure vessel overpressure conditions to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $31,799. This analysis case was used to model the benefit of phase II SAMA 49.

Improve SRV Design

This analysis case was used to evaluate the change in plant risk from improving the SRV design to increase the reliability of opening, thus increasing the likelihood that accident sequences could be mitigated using low pressure injection systems. A bounding analysis was performed by setting the probability of SRVs failing to open during RPV depressurization to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $194,378. This analysis case was used to model the benefit of phase II SAMA 50.

Self-Cooled ECCS Pump Seals

This analysis case was used to evaluate the change in plant risk from providing self-cooled ECCS pump seals to eliminate dependence on the component cooling water system. A bounding analysis was performed by setting the CDF contribution from sequences involving RHR pump failures to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $29,412. This analysis case was used to model the benefit of phase II SAMA 51.

Large Break LOCA

This analysis case was used to evaluate the change in plant risk from installing a digital large break LOCA protection system. A bounding analysis was performed by setting the large break LOCA initiator to zero in the level 1 PSA model, which resulted in an upper bound benefit of approximately $14,109. This analysis case was used to model the benefit of phase II SAMA 52.

Controlled Containment Venting

This analysis case was used to evaluate the change in plant risk from changing the design of the containment vent valves and procedure to establish a narrow pressure control band. This would prevent rapid containment depressurization when venting, thus avoiding adverse impact on the ability of the low pressure ECCS injection systems to take suction from the torus. A bounding analysis was performed by reducing the probability of the operator failing to recognize the need to vent the torus by a factor of three in the level 1 PSA model, which resulted in an upper bound benefit of approximately $137,237. This analysis case was used to model the benefit of phase II SAMA 53.

ECCS Low Pressure Interlock

This analysis case was used to evaluate the change in plant risk from installing a bypass switch to allow operator to bypass the ECCS low pressure interlock circuitry that inhibits opening of the RHR low pressure injection and core spray injection valves following sensor or logic failure. A bounding analysis was performed by setting the CDF contribution due to sensor failure, low pressure permissive logic failure, and miscalibration to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $21,761. This analysis case was used to model the benefit of phase II SAMA 54.

Improve the Reliability of SSW and RBCCW Pumps

This analysis case was used to evaluate the change in plant risk from providing a separate pump train to eliminate common cause failure of SSW and RBCCW pumps. A bounding analysis was performed by setting the CDF contribution due to common cause failures of SSW and RBCCW pumps to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $356,310. This analysis case was used to model the benefit of phase II SAMA 55.

Redundant DC Power Supplies to DTV Valves

This analysis case was used to evaluate the change in plant risk from installing additional fuses to two DTV valve control circuits to enable the DTV function. A bounding analysis was performed by setting the CDF contribution due to DC power supply failures to DTV valves AO-5042B and AO-5025 to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $220,639. This analysis case was used to model the benefit of phase II SAMA 56.

Proceduralize the Use of Diesel Fire Pump Hydroturbine

This analysis case was used to evaluate the change in plant risk from revising the procedure to allow use of hydroturbine if EDG X-107A or diesel driven fire water pump P-140 is unavailable. A bounding analysis was performed by setting the CDF contribution from the sequences involving a LOOP and failure of either EDG A or fuel oil transfer oil pump (P-141) to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $175,279. This analysis case was used to model the benefit of phase II SAMA 57.

Proceduralize Alignment of Bus B3 to Feed Bus B1 Loads or Bus B4 to Bus B2

This analysis case was used to evaluate the change in plant risk from providing a procedure to direct the operator to restore 480V MCCs B15 and B17 loads upon loss of 4.16kV bus A5 provided that 4.16kV bus A3 is available. The same is true for restoring 480V MCCs B14 and B18 loads upon loss of 4.16kV bus A6 provided that 4.16kV bus A4 is available. A bounding analysis was performed by setting the CDF contribution from the sequences involving a loss of the 4.16 kV bus A5 to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $190,797. This analysis case was used to model the benefit of phase II SAMA 58.

Redundant Path from Fire Water Pump Discharge to LPCI Loops A and B Cross-tie

This analysis case was used to evaluate the change in plant risk from installing a redundant path from fire protection water pump discharge to LPCI loops A and B cross-tie. A bounding analysis was performed by setting the CDF contribution from the sequences involving fire water into LPCI loops A and B cross-tie failure to zero in the level 1 PSA model. This resulted in an upper bound benefit of approximately $929,797. This analysis case was used to model the benefit of phase II SAMA 59.

E.2.4 Sensitivity Analyses

Two sensitivity analyses were conducted to gauge the impact of assumptions upon the analysis. The benefits estimated for each of these sensitivities are presented in Table E.2-2.

A description of each sensitivity case follows.

Sensitivity Case 1: Years Remaining Until End of Plant Life

The purpose of this sensitivity case was to investigate the sensitivity of assuming a 27-year period for remaining plant life (i.e. seven years on the original plant license plus the 20-year license renewal period). The 20-year license renewal period was used in the base case. The resultant monetary equivalent was calculated using 27 years remaining until end of facility life to investigate the impact on each analysis case. Changing this assumption does not cause any additional SAMAs to be cost-beneficial.

Sensitivity Case 2: Conservative Discount Rate

The purpose of this sensitivity case was to investigate the sensitivity of each analysis case to the discount rate. The discount rate of 7.0% used in the base case analyses is conservative relative to corporate practices. Nonetheless, a lower discount rate of 3.0% was assumed in this case to investigate the impact on each analysis case. Changing this assumption does not cause any additional SAMAs to be cost-beneficial.

E.2.5 References

E.2-1 Appendix D-Attachment F, Severe Accident Mitigation Alternatives Submittal Related to Licensing Renewal for the Edwin I. Hatch Nuclear Power Plant Units 1 and 2, March 2000.

E.2-2 U.S. Nuclear Regulatory Commission, NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, *Calvert Cliffs Nuclear Power Plant*, Supplement 1, February 1999.

E.2-3 General Electric Nuclear Energy, Technical Support Document for the ABWR, 25A5680, Revision 1, January 18, 1995.

E.2-4 Appendix E- Environmental Report, Appendix G, Severe Accident Mitigation Alternatives Submittal Related to Licensing Renewal for the Peach Bottom Nuclear Power Plant Units 2 and 3, July, 2001.

E.2-5 Appendix F, Severe Accident Mitigation Alternatives Analysis Submittal Related to Licensing Renewal for the Quad Cities Nuclear Power Plant Units 1 and 2, January 2003.

E.2-6 Appendix F, Severe Accident Mitigation Alternatives Analysis Submittal Related to Licensing Renewal for the Dresden Nuclear Power Plant Units 2 and 3, January 2003.

E.2-7 Appendix E-Attachment E, Severe Accident Mitigation Alternatives Submittal Related to Licensing Renewal for the Arkansas Nuclear One - Unit 2, October 2003.

E.2-8 Cost Estimate for Severe Accident Mitigation Design Alternatives, Limerick Generating Station for Philadelphia Electric Company, Bechtel Power Corporation, June 22, 1989.

E.2-9 U.S. Nuclear Regulatory Commission, NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, Volume 1, 5.35, Listing of SAMDAs considered for the Limerick Generating Station, May 1996.

E.2-10 U.S. Nuclear Regulatory Commission, NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants*, Volume 1, 5.36, Listing of SAMDAs considered for the Comanche Peak Steam Electric Station, May 1996.

E.2-11 Museler, W. J., (Tennessee Valley Authority) to NRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Severe Accident Mitigation Design Alternatives (SAMDAs)," letter dated October 7, 1994.

E.2-12 Nunn, D. E., (TVA) to NRC Document Control Desk, "Watts Bar Nuclear Plant (WBN) Units 1 and 2 - Severe Accident Mitigation Design Alternatives (SAMDA) - Response to Request for Additional Information (RAI) - (TAC Nos. M77222 and M77223)," letter dated October 7, 1994.

E.2-13 Liparulo, N. J., (Westinghouse Electric Corporation) to NRC Document Control Desk, "Submittal of Material Pertinent to the AP600 Design Certification Review," letter dated December 15, 1992.

E.2-14 U.S. Nuclear Regulatory Commission, NUREG-0498, *Final Environmental Statement related to the operation of Watts Bar Nuclear Plant, Units 1 and 2,* Supplement No. 1, April 1995.

E.2-15 U.S. Nuclear Regulatory Commission, NUREG-1560, *Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance*, Volume 2, December 1997.

E.2-16 U.S. Nuclear Regulatory Commission, NUREG/CR-5474, *Assessment of Candidate Accident Management Strategies*, March 1990.

E.2-17 Pilgrim Nuclear Power Station, Individual Plant Examination (IPE) Report, September 1992

E.2-18 Pilgrim Nuclear Power Station, Individual Plant Examination of External Events (IPEEE) Report, July 1994.

E.2-19 U.S. Nuclear Regulatory Commission, NUREG/BR-0184, *Regulatory Analysis Technical Evaluation Handbook*, January 1997.

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
Improvements Related to Accident Mitigation Containment Phenomena								
001	Install an independent method of suppression pool cooling.	SAMA would decrease the probability of loss of containment heat removal.	4.70%	4.60%	$43,639	$261,832	$5,800,000	Not cost effective
	Basis for Conclusion: The CDF contribution from loss of the torus cooling mode of RHR was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $5.8 million. Therefore, this SAMA is not cost effective for PNPS.							
002	Install a filtered containment vent to provide fission product scrubbing. Option 1: Gravel Bed Filter Option 2: Multiple Venturi Scrubber	SAMA would provide an alternate decay heat removal method for non-ATWS events, with fission product scrubbing.	0.00%	0.00%	$0	$0	$3,000,000	Not cost effective
	Basis for Conclusion: Successful torus venting accident progression source terms are reduced by a factor of 2 to reflect the additional filtered capability. The cost of implementing this SAMA at Peach Bottom was estimated to be $3 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
003	Install a containment vent large enough to remove ATWS decay heat.	Assuming that injection is available, this SAMA would provide alternate decay heat removal in an ATWS event.	0.50%	1.19%	$10,283	$61,701	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution from ATWS sequences associated with containment bypass were eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
004	Create a large concrete crucible with heat removal potential under the base mat to contain molten core debris.	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 base mat.	0.00%	48.62%	$436,759	$2,620,551	>$100 million	Not cost effective
	Basis for Conclusion: Containment failure due to core-concrete interactions (not including liner failures) was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at ANO-2 was estimated to be $100 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
005	Create a water-cooled rubble bed on the pedestal.	SAMA would contain molten core debris dropping on to the pedestal and would allow the debris to be cooled.	0.00%	48.62%	$436,759	$2,620,551	$19,000,000	Not cost effective
	Basis for Conclusion: Containment failure due to core-concrete interactions (not including liner failures) was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at ANO-2 was estimated to be $19 million. Therefore, this SAMA is not cost effective for PNPS.							
006	Provide modification for flooding the drywell head.	SAMA would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail.	0.00%	0.07%	$2,153	$12,915	>$1,000,000	Not cost effective
	Basis for Conclusion: Drywell head failures due to high temperature were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $1 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
007	Enhance fire protection system and SGTS hardware and procedures.	SAMA would improve fission product scrubbing in severe accidents.	0.00%	1.16%	$10,763	$64,577	>$2,500,000	Not cost effective
	Basis for Conclusion: Failure of the reactor building to contain releases was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2.5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
008	Create a core melt source reduction system.	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 from the vitrified compound would be facilitated, and concrete attack would not occur.	0.00%	48.62%	$436,759	$2,620,551	>$5,000,000	Not cost effective
	Basis for Conclusion: Containment failure due to core-concrete interactions (not including liner failures) was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
009	Install a passive containment spray system.	SAMA would decrease the probability of loss of containment heat removal.	5.05%	4.70%	$44,037	$264,219	$5,800,000	Not cost effective
	Basis for Conclusion: The CDF contribution from loss of the drywell spray mode of RHR was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $5.8 million. Therefore, this SAMA is not cost effective for PNPS.							
010	Strengthen primary and secondary containment.	SAMA would reduce the probability of containment over-pressurization failure.	0.00%	26.10%	$205,571	$1,233,428	$12,000,000	Not cost effective
	Basis for Conclusion: Energetic containment failure modes (DCH, steam explosion, late over-pressurization) were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities and at an ABWR was estimated to be $12 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
011	Increase the depth of the concrete base mat or use an alternative concrete material to ensure melt-through does not occur.	SAMA would prevent base mat melt-through.	0.00%	0.43%	$4,305	$25,831	>$5,000,000	Not cost effective
	Basis for Conclusion: Containment failure due to base mat melt-through was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
012	Provide a reactor vessel exterior cooling system.	SAMA would provide the potential to cool a molten core before it causes vessel failure, if the lower head could be submerged in water.	0.00%	0.22%	$3,229	$19,373	$2,500,000	Not cost effective
	Basis for Conclusion: The probability of vessel failure was modified to account for potential ex-vessel cooling of the vessel bottom head region to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $2.5 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
013	Construct a building connected to primary containment that is maintained at a vacuum.	SAMA would provide a method to depressurize containment and reduce fission product release.	0.00%	1.16%	$10,763	$64,577	>$2,000,000	Not cost effective
	Basis for Conclusion: Failure of the reactor building to contain releases was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2 million at Peach Bottom. Therefore, this SAMA is not cost effective for PNPS.							
014	2.g. Dedicated Suppression Pool Cooling	SAMA would decrease the probability of loss of containment heat removal.	4.70%	4.60%	$43,639	$261,832	$5,800,000	Not cost effective
	Basis for Conclusion: The CDF contribution from loss of the torus cooling mode of RHR was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $5.8 million. Therefore, this SAMA is not cost effective for PNPS.							
015	3.a. Create a larger volume in containment.	SAMA increases time before containment failure and increases time for recovery.	0.00%	26.10%	$205,571	$1,233,428	$8,000,000	Not cost effective
	Basis for Conclusion: Energetic containment failure modes (DCH, steam explosion, late over-pressurization) were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $8 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
016	3.b. Increase containment pressure capability (sufficient pressure to withstand severe accidents).	SAMA minimizes likelihood of large releases.	0.00%	26.10%	$205,571	$1,233,428	$12,000,000	Not cost effective
	Basis for Conclusion: Energetic containment failure modes (DCH, steam explosion, late over-pressurization) were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities and at an ABWR was estimated to be $12 million. Therefore, this SAMA is not cost effective for PNPS.							
017	3.c. Install improved vacuum breakers (redundant valves in each line).	This SAMA addresses the reliability of a vacuum breaker to reseat following a successful opening.	0.00%	0.00%	$0	$0	>$1,000,000	Not cost effective
	Basis for Conclusion: Vacuum breaker failures and suppression pool scrubbing failures were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $1 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
018	3.d. Increase the temperature margin for seals.	This SAMA would reduce the potential for containment failure under adverse conditions.	0.00%	0.07%	$2,153	$12,915	$12,000,000	Not cost effective
	Basis for Conclusion: Containment failure due to high temperature drywell seal failure was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities and at an ABWR were estimated to be $12 million and was judged to exceed the attainable benefit, even without a detailed cost estimate. Therefore, this SAMA is not cost effective for PNPS.							
019	5.b/c. Install a filtered vent	SAMA would provide an alternate decay heat removal method for non-ATWS events, with fission product scrubbing.	0.00%	0.00%	$0	$0	$3,000,000	Not cost effective
	Basis for Conclusion: Successful torus venting accident progressions source terms are reduced by a factor of 2 to reflect the additional filtered capability. The cost of implementing this SAMA at Peach Bottom was estimated to be $3 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
020	7.a. Provide a method of drywell head flooding.	SAMA would provide intentional flooding of the upper drywell head such that if high drywell temperatures occurred, the drywell head seal would not fail.	0.00%	0.07%	$2,153	$12,915	>$1,000,000	Not cost effective
	Basis for Conclusion: Drywell head failures due to high temperature were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $1 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
021	13.a. Use alternate method of reactor building spray.	This SAMA provides the capability to use firewater sprays in the reactor building to mitigate release of fission products into the reactor building following an accident.	0.00%	1.16%	$10,763	$64,577	>$2,500,000	Not cost effective
	Basis for Conclusion: Failure of the reactor building to contain releases was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2.5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
022	14.a. Provide a means of flooding the rubble bed.	SAMA would allow the debris to be cooled.	0.00%	22.48%	$204,495	$1,226,971	$2,500,000	Not cost effective
	Basis for Conclusion: The probabilities of wet core concrete interactions were substituted for dry core concrete interactions to assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $2.5 million. Therefore, this SAMA is not cost effective for PNPS.							
023	14.b. Install a reactor cavity flooding system.	SAMA would enhance debris coolability, reduce core concrete interaction, and provide fission product scrubbing.	0.00%	48.62%	$436,759	$2,620,551	$8,750,000	Not cost effective
	Basis for Conclusion: Containment failure due to core-concrete interactions (not including liner failures) was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at ANO-2 was estimated to be $8.75 million. Therefore, this SAMA is not cost effective for PNPS.							
024	Add ribbing to the containment shell.	This SAMA would reduce the chance of containment buckling under reverse pressure loading.	0.00%	26.10%	$205,571	$1,233,428	$12,000,000	Not cost effective
	Basis for Conclusion: Energetic containment failure modes (DCH, steam explosion, late over-pressurization) were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities and at an ABWR was estimated to be $12 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
Improvements Related to Enhanced AC/DC Reliability/Availability								
025	Provide additional DC battery capacity.	SAMA would ensure longer battery capability during an SBO, which would extend HPCI/RCIC operability and allow more time for AC power recovery.	1.39%	2.79%	$24,393	$146,356	$500,000	Not cost effective
	Basis for Conclusion: The time available to recover offsite power before HPCI and RCIC are lost was changed from 14 hours to 24 hours during SBO scenarios to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $500,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
026	Use fuel cells instead of lead-acid batteries.	SAMA would extend DC power availability in an SBO, which would extend HPCI/RCIC operability and allow more time for AC power recovery.	1.39%	2.79%	$24,393	$146,356	>$2,000,000	Not cost effective
	Basis for Conclusion: The time available to recover offsite power before HPCI and RCIC are lost was changed from 14 hours to 24 hours during SBO scenarios to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
027	Modification for Improving DC Bus Reliability	SAMA would increase reliability of AC power and injection capability.	4.65%	1.91%	$19,761	$118,568	$500,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to loss of DC buses D16 and D17 was eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $500,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
028	2.i. Provide 16-hour SBO injection.	SAMA includes improved capability to cope with longer SBO scenarios.	1.39%	2.79%	$24,393	$146,356	$500,000	Not cost effective
	Basis for Conclusion: The time available to recover offsite power before HPCI and RCIC are lost was changed from 14 hours to 24 hours during SBO scenarios to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $500,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
029	9.b. Provide an alternate pump power source.	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.	2.22%	5.06%	$44,281	$265,687	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the SBO diesel was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
030	9.g. Enhance procedures to make use of AC bus cross-ties.	SAMA would provide increased reliability of AC power system and reduce core damage and release frequencies.	11.10%	8.47%	$78,902	$473,410	$146,120	Retain
	Basis for Conclusion: The CDF contribution due to loss of MCCs B17, B18, and B15 was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $146,120 by engineering judgment.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
031	10.a. Add a dedicated DC power supply.	This SAMA addresses the use of a diverse DC power system such as an additional battery or fuel cell for the purpose of providing motive power to certain components (e.g., RCIC).	24.3%	16.16%	$150,504	$903,025	$3,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to loss of DC Bus 'B' was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $3 million. Therefore, this SAMA is not cost effective for PNPS.							
032	10.b. Install additional batteries or divisions.	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).	24.3%	16.16%	$150,504	$903,025	$3,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to loss of DC Bus 'B' was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be $3 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
033	10.c. Install fuel cells.	SAMA would extend DC power availability in an SBO, which would extend HPCI/RCIC operability and allow more time for AC power recovery.	1.39%	2.79%	$24,393	$146,356	>$2,000,000	Not cost effective
	Basis for Conclusion: The time available to recover offsite power before HPCI and RCIC are lost was changed from 14 hours to 24 hours during SBO scenarios to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
034	10.d. Enhance procedures to make use of DC bus cross-ties.	This SAMA would improve DC power availability.	4.65%	1.91%	$19,761	$118,568	$13,000	Retain
	Basis for Conclusion: The CDF contribution due to loss of DC buses D16 and D17 was eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $13,000 by engineering judgment.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
035	10.e. Extended SBO provisions.	SAMA would extend DC power availability in an SBO, which would extend HPCI/RCIC operability and allow more time for AC power recovery.	1.39%	2.79%	$24,393	$146,356	$500,000	Not cost effective
	Basis for Conclusion: The time available to recover offsite power before HPCI and RCIC are lost was changed from 14 hours to 24 hours during SBO scenarios to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $500,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
Improvements in Identifying and Mitigating Containment Bypass								
036	Locate RHR inside containment.	SAMA would prevent ISLOCA outside containment.	0.33%	0.21%	$2,749	$16,497	>$500,000	Not cost effective
	Basis for Conclusion: RHR ISLOCA accident sequences were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Quad Cities was estimated to be greater than $500.000. Therefore, this SAMA is not cost effective for PNPS.							
037	Increase frequency of valve leak testing.	SAMA could reduce ISLOCA frequency.	0.54%	0.38%	$4,025	$24,148	$100,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to ISLOCA was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $100,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
038	8.e. Improve MSIV design.	This SAMA would decrease the likelihood of containment bypass scenarios.	0.00%	0.00%	$0	$0	>$2,000,000	Not cost effective
	Basis for Conclusion: Containment bypass failure due to MSIV leakage was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
Improvements Related to Core Cooling System								
039	Install an independent diesel for the CST makeup pumps.	SAMA would allow continued inventory in CST during an SBO.	0.00%	0.00%	$0	$0	$135,000	Not cost effective
	Basis for Conclusion: As currently modeled, if CST water level is low, swapping HPCI/RCIC suction from the CST to the torus allows continued HPCI/RCIC injection. Therefore, the failure to switchover from CST to torus was eliminated to conservatively assess the benefit of this SAMA on CDF. The cost of implementing this SAMA was estimated to be $135,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
040	Provide an additional high pressure injection pump with independent diesel.	SAMA would reduce frequency of core melt from small LOCA and SBO sequences.	3.15%	1.97%	$18,369	$110,212	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the HPCI system was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
041	Install independent AC high pressure injection system.	SAMA would allow makeup capabilities during transients, small LOCAs, and SBOs.	3.15%	1.97%	$18,369	$110,212	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the HPCI system was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
042	2.a. Install a passive high pressure system.	SAMA would improve prevention of core melt sequences by providing additional high pressure capability to remove decay heat through an isolation condenser type system.	3.15%	1.97%	$18,369	$110,212	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the HPCI system was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2 million at Peach Bottom. Therefore, this SAMA is not cost effective for PNPS.							
043	2.d. Improved high pressure systems	SAMA will improve prevention of core melt sequences by improving reliability of high pressure capability to remove decay heat.	2.11%	1.43%	$12,671	$76,025	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution from reducing the HPCI system failure probability by a factor of 3 was estimated to bound the potential impact of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2 million at Peach Bottom. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
044	2.e. Install an additional active high pressure system.	SAMA will improve reliability of high-pressure decay heat removal by adding an additional system.	3.15%	1.97%	$18,369	$110,212	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the HPCI system was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							
045	8.c. Add a diverse injection system.	SAMA will improve prevention of core melt sequences by providing additional injection capabilities.	3.15%	1.97%	$18,369	$110,212	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to failure of the HPCI system was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
Improvements Related to ATWS Mitigation								
046	Increase SRV reseat reliability.	SAMA addresses the risk associated with dilution of boron caused by the failure of the SRVs to reseat after SLC injection.	1.51%	0.92%	$10,600	$63,599	$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to stuck open relief valves was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $2 million at Peach Bottom. Therefore, this SAMA is not cost effective for PNPS.							
047	11.a. Install an ATWS sized vent.	This SAMA would provide the ability to remove reactor heat from ATWS events.	0.50%	1.19%	$10,283	$61,701	>$2,000,000	Not cost effective
	Basis for Conclusion: The CDF contribution from ATWS sequences associated with containment bypass were eliminated to conservatively assess the benefit of this SAMA. The cost of implementing of this SAMA at Peach Bottom was estimated to be greater than $2 million. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
048	Diversify explosive valve operation.	An alternate means of opening a pathway to the RPV for SLC system injection would improve the success probability for reactor shutdown.	0.00%	0.02%	$2,153	$12,915	>$200,000	Not cost effective
	Basis for Conclusion: Common cause failure of SLC explosive valves was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $200,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							
Other Improvements								
049	Increase the reliability of SRVs by adding signals to open them automatically.	SAMA reduces the consequences of medium break LOCAs.	0.73%	0.60%	$5,300	$31,799	>$1,500,000	Not cost effective
	Basis for Conclusion: The CDF contribution from SRVs failing to open in medium LOCA sequences was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $1.5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
050	8.e. Improve SRV design.	This SAMA would improve SRV reliability thus increasing the likelihood that sequences could be mitigated using low-pressure heat removal.	4.81%	3.51%	$32,396	$194,378	>$2,000,000	Not cost effective
	Basis for Conclusion: The probability of SRV failure to open for vessel depressurization was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $2 million at Peach Bottom. Therefore, this SAMA is not cost effective for PNPS.							
051	Provide self-cooled ECCS pump seals.	SAMA would eliminate ECCS dependency on the component cooling water system.	0.47%	0.55%	$4,902	$29,412	>$200,000	Not cost effective
	Basis for Conclusion: The CDF contribution from sequences involving RHR pump failures was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $200,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
052	Provide digital large break LOCA protection.	Upgrade plant instrumentation and logic to improve the capability to identify symptoms/precursors of a large break LOCA (a leak before break).	0.07%	0.01%	$2,352	$14,109	>$100,000	Not cost effective
	Basis for Conclusion: The CDF contribution due to large break LOCA was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $100,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
Improvements Related to IPE, IPE Update& IPEEE Insights								
053	Control containment venting within a narrow band of pressure	This SAMA would establish a narrow pressure control band to prevent rapid containment depressurization when venting is implemented thus avoiding adverse impact on the low pressure ECCS injection systems taking suction from the torus.	3.61%	2.24%	$22,873	$137,237	$300,000	Not cost effective
	Basis for Conclusion: The probability of the operator failing to recognize the need to vent the torus was reduced by a factor of 3 to conservatively assess the benefit of this SAMA on CDF. The cost of implementing this SAMA was estimated to be $300,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
054	Install a bypass switch to bypass the low reactor pressure interlocks of LPCI or core spray injection valves	This SAMA would reduce the core damage frequency contribution from the transients with stuck open SRVs or LOCAs cases. Core Spray and LPCI injection valves require a low permissive signal from the same two sensors to open the valves for RPV injection.	0.28%	0.33%	$3,627	$21,761	$1,000,000	Not cost effective
	Basis for Conclusion: The probability of the ECCS low-pressure permissive failing was eliminated to conservatively assess the benefit of this SAMA on CDF. The cost of implementing this SAMA at Dresden was estimated to be $1 million. Therefore, this SAMA is not cost effective for PNPS.							
055	Increase the reliability of SSW and RBCCW pumps.	This SAMA would reduce common cause dependencies from SSW and RBCCW systems and thus reduce plant risk.	4.37%	6.63%	$59,385	$356,310	>$5 million	Not cost effective
	Basis for Conclusion: The CDF contribution from sequences involving common cause failures of SSW and RBCCW was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be greater than $5 million by engineering judgment. Therefore, this SAMA is not cost effective for PNPS.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
056	Provide redundant DC power supplies to DTV valves.	This SAMA would improve reliability of the DTV valves and enhance containment heat removal capability.	8.81%	3.51%	$36,773	$220,639	$112,400	Retain
	Basis for Conclusion: The CDF contribution from sequences involving DC power supply failures to the DTV valves was eliminated to conservatively assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $112,400 by engineering judgment.							
057	Proceduralize use of the diesel fire pump hydro turbine in the event of EDG A failure or unavailability.	This SAMA would increase capability to provide makeup to the fire pump day tank to allow continued operation of the diesel fire pump, without dependence on electrical power.	2.25%	3.14%	$29,213	$175,279	$26,000	Retain
	Basis for Conclusion: The CDF contribution from sequences involving a LOOP and failure of either EDG A, or the EDG A fuel oil transfer oil pump, was eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $26,000 by engineering judgment.							

Table E.2-1
Summary of Phase II SAMA Candidates Considered in Cost-Benefit Evaluation (Continued)

Phase II SAMA ID	SAMA	Result of Potential Enhancement	CDF Reduction	Off-Site Dose Reduction	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Conclusion
058	Proceduralize the operator action to feed B1 loads via B3 When A5 is unavailable post-trip. Similarly, feed B2 loads via B4 when A6 is unavailable post trip.	This SAMA would provide the direction to restore B15 and B17 loads upon loss of A5 initiating events as long as A3 is available. Additionally, it would provide the direction to restore B14 and B18 loads upon loss of A6 initiating events as long as A4 is available.	4.92%	3.14%	$31,799	$190,797	$50,000	Retain
	Basis for Conclusion: The CDF contribution from sequences involving loss of 4160VAC safeguard bus A5 was conservatively eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $50,000 by engineering judgment.							
059	Provide redundant path from fire protection pump discharge to LPCI loops A and B cross-tie.	This SAMA would enhance the availability and reliability of the firewater cross-tie to LPCI loops A and B for reactor vessel injection and drywell spray.	8.77%	17.19%	$154,966	$929,797	$1,956,000	Not cost effective
	Basis for Conclusion: The CDF contribution from sequences involving firewater injection failures was conservatively eliminated to assess the benefit of this SAMA. The cost of implementing this SAMA was estimated to be $1,956,000 by engineering judgment. Therefore, this SAMA is not cost effective for PNPS							

Table E.2-2
Sensitivity Analysis Results

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
1	Install an independent method of suppression pool cooling.	$43,639	$261,832	$5,800,000	$50,320	$301,920	$59,355	$356,129
2	Install a filtered containment vent to provide fission product scrubbing. Option 1: Gravel Bed Filter Option 2: Multiple Venturi Scrubber	$0	$0	$3,000,000	$0	$0	$0	$0
3	Install a containment vent large enough to remove ATWS decay heat.	$10,283	$61,701	>$2,000,000	$11,702	$70,211	$14,207	$85,244
4	Create a large concrete crucible with heat removal potential under the basemat to contain molten core debris.	$436,759	$2,620,551	>$100 million	$492,136	$2,952,813	$610,307	$3,661,845
5	Create a water-cooled rubble bed on the pedestal.	$436,759	$2,620,551	$19,000,000	$498,057	$2,988,339	$610,307	$3,661,845

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Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
6	Provide modification for flooding the drywell head	$2,153	$12,915	>$1,000,000	$2,425	$14,551	$3,008	$18,048
7	Enhance fire protection system and/or SGTS hardware and procedures.	$10,763	$64,577	>$2,500,000	$12,127	$72,764	$15,040	$90,238
8	Create a core melt source reduction system.	$436,759	$2,620,551	>$5,000,000	$498,057	$2,988,339	$610,307	$3,661,845
9	Install a passive containment spray system.	$44,037	$264,219	$5,800,000	$50,845	$305,069	$59,803	$358,816
10	Strengthen primary/ secondary containment.	$205,571	$1,233,428	$12,000,000	$231,636	$1,389,815	$287,257	$1,723,540
11	Increase the depth of the concrete basemat or use an alternative concrete material to ensure melt-through does not occur	$4,305	$25,831	>$5,000,000	$4,851	$29,105	$6,016	$36,095
12	Provide a reactor vessel exterior cooling system (see #7)	$3,229	$19,373	$2,500,000	$3,638	$21,828	$4,512	$27,071

Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
13	Construct a building to be connected to primary/ secondary containment that is maintained at a vacuum	$10,763	$64,577	>$2,000,000	$12,273	$73,640	$15,040	$90,238
14	2.g. Dedicated Suppression Pool Cooling	$43,639	$261,832	$5,800,000	$51,067	$306,400	$59,355	$356,129
15	3.a. Create a larger volume in containment.	$205,571	$1,233,428	$8,000,000	$234,423	$1,406,537	$287,257	$1,723,540
16	3.b. Increase containment pressure capability (sufficient pressure to withstand severe accidents).	$205,571	$1,233,428	$12,000,000	$234,423	$1,406,537	$287,257	$1,723,540
17	3.c. Install improved vacuum breakers (redundant valves in each line).	$0	$0	>$1,000,000	$0	$0	$0	$0
18	3.d. Increase the temperature margin for seals.	$2,153	$12,915	$12,000,000	$2,455	$14,728	$3,008	$18,048
19	5.b/c. Install a filtered vent	$0	$0	$3,000,000	$0	$0	$0	$0

Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
20	7.a. Provide a method of drywell head flooding.	$2,153	$12,915	>$1,000,000	$2,455	$14,728	$3,008	$18,048
21	13.a. Use alternate method of reactor building spray.	$10,763	$64,577	>$2,500,000	$12,273	$73,640	$15,040	$90,238
22	14.a. Provide a means of flooding the rubble bed.	$204,495	$1,226,971	$2,500,000	$230,423	$1,382,539	$285,753	$1,714,516
23	14.b. Install a reactor cavity flooding system.	$436,759	$2,620,551	$8,750,000	$498,057	$2,988,339	$610,307	$3,661,845
24	Add ribbing to the containment shell.	$205,571	$1,233,428	$12,000,000	$234,423	$1,406,537	$287,257	$1,723,540
25	Provide additional DC battery capacity.	$24,393	$146,356	$500,000	$27,830	$166,978	$33,598	$201,588
26	Use fuel cells instead of lead-acid batteries.	$24,393	$146,356	>$2,000,000	$28,207	$169,242	$33,598	$201,588
27	Modification for Improving DC Bus Reliability	$19,761	$118,568	$500,000	$23,377	$140,262	$26,044	$156,263
28	2.i. Provide 16-hour SBO injection.	$24,393	$146,356	$500,000	$28,207	$169,242	$33,598	$201,588

Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
29	9.b. Provide an alternate pump power source.	$44,281	$265,687	>$2,000,000	$50,546	$303,278	$60,956	$365,738
30	9.g. AC Bus Cross-Ties	$78,902	$473,410	$146,120	$91,662	$549,972	$106,357	$638,142
31	10.a. Add a dedicated DC power supply.	$150,504	$903,025	$3,000,000	$178,405	$1,070,432	$201,864	$1,211,183
32	10.b. Install additional batteries or divisions.	$150,504	$903,025	$3,000,000	$178,405	$1,070,432	$201,864	$1,211,183
33	10.c. Install fuel cells.	$24,393	$146,356	>$2,000,000	$28,207	$169,242	$33,598	$201,588
34	10.d. DC Cross-Ties	$19,761	$118,568	$13,000	$23,377	$140,262	$26,044	$156,263
35	10.e. Extended SBO provisions.	$24,393	$146,356	$500,000	$28,207	$169,242	$33,598	$201,588
36	Locate RHR inside containment.	$2,749	$16,497	>$500,000	$3,213	$19,276	$3,680	$22,077
37	Increase frequency of valve leak testing.	$4,025	$24,148	$100,000	$4,688	$28,127	$5,407	$32,444
38	8.e. Improve MSIV design.	$0	$0	>$2,000,000	$0	$0	$0	$0

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Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
39	Install an independent diesel for the CST makeup pumps.	$0	$0	$135,000	$0	$0	$0	$0
40	Provide an additional high pressure injection pump with independent diesel.	$18,369	$110,212	>$2,000,000	$21,540	$129,238	$24,477	$146,860
41	Install independent AC high pressure injection system.	$18,369	$110,212	>$2,000,000	$21,902	$131,415	$24,477	$146,860
42	2.a. Install a passive high pressure system.	$18,369	$110,212	>$2,000,000	$21,902	$131,415	$24,477	$146,860
43	2.d. Improved high pressure systems	$12,671	$76,025	>$2,000,000	$14,851	$89,109	$16,894	$101,363
44	2.e. Install an additional active high pressure system.	$18,369	$110,212	>$2,000,000	$21,902	$131,415	$24,477	$146,860
45	8.c. Add a diverse injection system.	$18,369	$110,212	>$2,000,000	$21,902	$131,415	$24,477	$146,860
46	Increase SRV reseat reliability.	$10,600	$63,599	$2,000,000	$12,326	$73,958	$14,270	$85,623
47	11.a. Install an ATWS sized vent.	$10,283	$61,701	>$2,000,000	$11,857	$71,142	$14,207	$85,244

Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
48	Diversify explosive valve operation.	$2,153	$12,915	>$200,000	$2,425	$14,551	$3,008	$18,048
49	Increase the reliability of SRVs by adding signals to open them automatically.	$5,300	$31,799	>$1,500,000	$6,163	$36,978	$7,135	$42,811
50	8.e. Improve SRV design.	$32,396	$194,378	>$2,000,000	$37,767	$226,602	$43,483	$260,897
51	Provide self-cooled ECCS pump seals.	$4,902	$29,412	>$200,000	$5,638	$33,829	$6,687	$40,125
52	Provide digital large break LOCA protection.	$2,352	$14,109	>$100,000	$2,688	$16,126	$3,232	$19,391
53	Control containment venting within a narrow band of pressure	$22,873	$137,237	$300,000	$26,653	$159,919	$30,716	$184,299
54	Install a bypass switch to bypass the low reactor pressure interlocks of LPCI or core spray injection valves.	$3,627	$21,761	$1,000,000	$4,163	$24,978	$4,960	$29,758
55	Improve SSW System and RBCCW pump recovery.	$59,385	$356,310	>$5 million	$67,986	$407,918	$81,467	$488,799

Table E.2-2
Sensitivity Analysis Results (Continued)

Phase II SAMA ID	SAMA	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Cost	Estimated Benefit	Upper Bound Estimated Benefit	Estimated Benefit	Upper Bound Estimated Benefit
		Base Line	Base Line		Sensitivity Case 1	Sensitivity Case 1	Sensitivity Case 2	Sensitivity Case 2
56	Provide redundant DC power supplies to DTV valves.	$36,773	$220,639	$112,400	$43,541	$261,247	$48,408	$290,449
57	Proceduralize the use of diesel fire pump hydroturbine in the event of EDG A failure or unavailability.	$29,213	$175,279	$26,000	$33,568	$201,406	$39,901	$239,406
58	Proceduralize the operator action to feed B1 loads via B3 When A5 is unavailable post-trip.	$31,799	$190,797	$50,000	$36,980	$221,878	$42,811	$256,868
59	Provide redundant path from fire protection pump discharge to LPCI loops A and B cross-tie.	$154,966	$929,797	$1,956,000	$176,682	$1,060,091	$213,620	$1,281,720