



FirstEnergy Nuclear Operating Company

**Peter P. Sena III**  
Site Vice President

724-682-5234  
Fax: 724-643-8069

March 7, 2008  
L-08-081

10 CFR 54

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

**SUBJECT:**

Beaver Valley Power Station, Unit Nos. 1 and 2  
BV-1 Docket No. 50-334, License No. DPR-66  
BV-2 Docket No. 50-412, License No. NPF-73  
License Renewal Application Amendment 2: Reply to Request for Additional Information  
Regarding Severe Accident Mitigation Alternatives for Beaver Valley Power Station  
Units 1 and 2 License Renewal

Reference 1 provided the FirstEnergy Nuclear Operating Company (FENOC) License Renewal Application for the Beaver Valley Power Station (BVPS). Reference 2 requested additional information regarding Severe Accident Mitigation Alternatives (SAMA) analyses described in Appendix E, "Environmental Report," of the BVPS License Renewal Application. This letter provides Amendment 2 to the BVPS License Renewal Application relative to the SAMA analyses in the Environmental Report, and the FENOC reply to the U.S. Nuclear Regulatory Commission (NRC) request for additional information.

The Attachment provides the FENOC reply to the NRC request for additional information (RAI) regarding SAMA for BVPS license renewal.

Enclosure A provides Scientech Calculation 17676-0002, "Beaver Valley Power Station MACCS2 Input Data."

Enclosure B provides tabulated revisions to the SAMA information for the BVPS License Renewal Application, Appendix E – Environmental Report, based on the FENOC reply to the NRC request for additional information.

There are no regulatory commitments contained in this letter. If there are any questions or if additional information is required, please contact Mr. Clifford I. Custer, Fleet License Renewal Project Manager, at 724-682-7139.

A108  
NRR

I declare under penalty of perjury that the foregoing is true and correct. Executed on March 7, 2008.

Sincerely,



Peter P. Sena III

References:

1. FENOC Letter L-07-113, "License Renewal Application," dated August 27, 2007.
2. NRC Letter, "Request for Additional Information Regarding Severe Accident Mitigation Alternatives for Beaver Valley Power Station Units 1 and 2 License Renewal (TAC Nos. MD6595 and MD6596)," January 28, 2008.

Attachment:

Reply to Request for Additional Information Regarding the Analysis of Severe Accident Mitigation Alternatives (SAMAs) for Beaver Valley Units 1 and 2 License Renewal

Enclosures:

- A. Scientech Calculation 17676-0002, "Beaver Valley Power Station MACCS2 Input Data," Revision 3, August 17, 2007
- B. Tabulation of Revisions to the BVPS License Renewal Application, Appendix E, "Applicant's Environmental Report – Operating License Renewal Stage," Attachment C, "Severe Accident Mitigation Alternatives"

cc: Mr. K. L. Howard, Project Manager  
Mr. S. J. Collins, NRC Region I Administrator

cc: w/o Attachment or Enclosures  
Dr. P. T. Kuo, Director, Division of License Renewal  
Mr. D. L. Werkheiser, NRC Senior Resident Inspector  
Ms. N. S. Morgan, NRR Project Manager  
Mr. D. J. Allard, Director BRP/DEP  
Mr. L. E. Ryan (BRP/DEP)

ATTACHMENT  
L-08-081

Reply to Request for Additional Information Regarding the Analysis of  
Severe Accident Mitigation Alternatives (SAMAs) for  
Beaver Valley Units 1 and 2 License Renewal  
Page 1 of 106

**Question SAMA-1**

**Provide the following information regarding the Probabilistic Risk Assessment (PRA) models used for the SAMA analysis (for both units unless otherwise specified):**

- a. The list of dominant contributors to core damage frequency (CDF) only accounts for 90 percent of the internal events CDF. Provide the complete CDF breakdown of the remaining initiating events.**
- b. Clarify whether anticipated transient without scram (ATWS) events are modeled in the external event analysis. Provide the ATWS and station blackout (SBO) CDF for both internal and external event initiators.**
- c. Provide a discussion on the loss of containment instrument air. Identify any differences in plant features or PRA models/assumptions that cause the loss of containment instrument air initiator (ICX) to be a larger contributor in Unit 2 than Unit 1.**
- d. Significant CDF reduction was achieved as a result of Revision 3 update for Unit 1, i.e., a reduction in internal event CDF from 6.24E-5 per year in Revision 2 to 7.45E-6 per year in Revision 3. Discuss the reasons for the reduction in CDF and the extent to which these changes were considered in the Westinghouse Owner's Group peer review.**
- e. It is indicated that the heat up rates for the switchgear rooms is slower than what was assumed during the Individual Plant Examination (IPE). The heatup times are now about 5 hours in Unit 1 (per discussion of SAMA 181 in Table 6-1) and more than 24 hours in Unit 2 (per discussion in Section 3.1.1.2). Describe the plant features or modeling that causes the difference in heat up rates between the units.**
- f. Discuss the peer reviews performed (internal and/or external to FENOC) and quality controls applied to the external event models, to the Level 2 PRA models, and to the internal event PRA revisions subsequent to the Westinghouse Owner's Group and Nuclear Energy Institute peer reviews (i.e., Revisions 3 and 4 for Unit 1 and Revisions 3B and 4 for Unit 2.)**
- g. Identify the major shared systems and components between the two units. Discuss how their risk contributions are accounted for in the PRA.**

**h. Provide the CDF for internal floods and provide a breakdown and summary of the top flood scenarios.**

RESPONSE SAMA-1.a

**a. The list of dominant contributors to core damage frequency (CDF) only accounts for 90 percent of the internal events CDF. Provide the complete CDF breakdown of the remaining initiating events.**

Unit 1:

The Unit 1 CDF breakdown of the remaining 78 internal initiating events (including internal floods) that were not provided in BVPS License Renewal Application (LRA), Appendix E – Environmental Report (ER), Attachment C-1, page C.1-10, Table 3.1.1.1-1, “BV1REV4 Dominant Initiating Event Contribution to Internal Core Damage,” are presented in Table 1.A-1, shown below.

**Table 1.A-1**

BV1REV4 Remaining Initiating Event Contribution to Internal Core Damage Frequency

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
VSX	Interfacing Systems LOCA (V-Sequence)	9.84E-06	2.05E-08	0.5%	90.6%
PABF2A	PAB Flood at EI 735 Train A - Not Isolated	2.54E-05	2.02E-08	0.5%	91.1%
PABF2B	PAB Flood at EI 735 Train B - Not Isolated	2.54E-05	2.00E-08	0.5%	91.6%
SGTRA	Steam Generator A Tube Rupture	6.97E-04	1.88E-08	0.5%	92.1%
SGTRB	Steam Generator B Tube Rupture	6.97E-04	1.84E-08	0.5%	92.6%
SGTRC	Steam Generator C Tube Rupture	6.97E-04	1.84E-08	0.5%	93.0%
PABF3A	PAB Flood at EI 722 Train A - Early Isolation	3.89E-04	1.79E-08	0.5%	93.5%
PABF1A	PAB Flood at EI 735 River Water Train A - Isolated	3.84E-04	1.78E-08	0.5%	93.9%
PABF1B	PAB Flood at EI 735 River Water Train B - Isolated	3.84E-04	1.67E-08	0.4%	94.4%
PABF3B	PAB Flood at EI 722 Train B - Early Isolation	3.89E-04	1.56E-08	0.4%	94.8%
IWX	Loss of Vital Bus II (White)	5.68E-03	1.51E-08	0.4%	95.1%
IRX	Loss of Vital Bus I (Red)	5.68E-03	1.49E-08	0.4%	95.5%
LB1A	Loss of Normal 4KV Bus 1A	3.51E-03	1.41E-08	0.4%	95.9%
TLMFWA	Total Loss of Main Feedwater - ATWS	4.14E-02	1.29E-08	0.3%	96.2%

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
LOSPY	Loss of Offsite Power - Switchyard Centered	7.74E-03	1.27E-08	0.3%	96.5%
LB1D	Loss Of Normal 4KV Bus 1D	3.51E-03	1.02E-08	0.3%	96.8%
TBFL	Turbine Building Flood	7.71E-03	9.22E-09	0.2%	97.0%
TTRIPA	Turbine Trip - ATWS	6.52E-01	7.51E-09	0.2%	97.2%
ICX	Loss of Containment Instrument Air	1.05E-02	6.19E-09	0.2%	97.4%
BPXA	Loss of Emergency 4160V AC Purple - ATWS	1.78E-02	5.70E-09	0.1%	97.5%
AOXA	Loss of Emergency 4160V AC Orange - ATWS	1.78E-02	5.70E-09	0.1%	97.6%
LLOCAA	Large Loss of Coolant Accident in Loop A	2.40E-06	5.14E-09	0.1%	97.8%
LLOCAB	Large Loss of Coolant Accident in Loop B	2.40E-06	5.14E-09	0.1%	97.9%
LLOCAC	Large Loss of Coolant Accident in Loop C	2.40E-06	5.14E-09	0.1%	98.0%
AMSIV	Closure of All MSIV's	1.33E-02	5.06E-09	0.1%	98.2%
LOSPW	Loss of Offsite Power - Severe Weather Related	2.98E-03	4.87E-09	0.1%	98.3%
IBXA	Loss of Vital Bus III (Blue) - ATWS	5.68E-03	4.67E-09	0.1%	98.4%
SLBD	Steam Line Break Outside Containment	4.41E-03	4.17E-09	0.1%	98.5%
DOXA	Loss of Emergency 125V DC Orange - ATWS	4.80E-03	3.83E-09	0.1%	98.6%
LOSPP	Loss of Offsite Power - Plant Centered	2.30E-03	3.75E-09	0.1%	98.7%
LOSPGA	Loss of Offsite Power - Grid Centered (ATWS)	1.34E-02	3.69E-09	0.1%	98.8%
IYXA	Loss of Vital Bus IV (Yellow) - ATWS	5.70E-03	3.58E-09	0.1%	98.9%
CPEXC	Core Power Excursion	1.13E-02	3.28E-09	0.1%	99.0%
IBX	Loss of Vital Bus III (Blue)	5.68E-03	2.86E-09	0.1%	99.0%
IYX	Loss of Vital Bus IV (Yellow)	5.70E-03	2.84E-09	0.1%	99.1%
DPXA	Loss of Emergency 125V DC Purple - ATWS	4.80E-03	2.67E-09	0.1%	99.2%
SGTRBA	Steam Generator B Tube Rupture - ATWS	6.97E-04	2.51E-09	0.1%	99.2%
SGTRAA	Steam Generator A Tube Rupture - ATWS	6.97E-04	2.51E-09	0.1%	99.3%
SGTRCA	Steam Generator C Tube Rupture - ATWS	6.97E-04	2.51E-09	0.1%	99.4%
SLBI	Steam Line Break Inside Containment	8.31E-04	2.38E-09	0.1%	99.4%
LOSPYA	Loss of Offsite Power - Switchyard Centered (ATWS)	7.74E-03	2.13E-09	0.1%	99.5%
PABF4A	PAB Flood at EI 722 Train A - Late Isolation	3.32E-05	1.53E-09	0.04%	99.5%

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
SLBDA	Steam Line Break Outside Containment - ATWS	4.41E-03	1.46E-09	0.04%	99.6%
CCX	Loss of Reactor Plant Component Cooling Water	2.46E-03	1.44E-09	0.04%	99.6%
PABF4B	PAB Flood at EI 722 Train B - Late Isolation	3.32E-05	1.43E-09	0.04%	99.6%
ISFLA	Intake Structure Flood in Cubicle A	9.01E-04	1.38E-09	0.03%	99.7%
LCVA	Loss of Condenser Vacuum - ATWS	1.16E-01	1.33E-09	0.03%	99.7%
IAX	Loss of Station Instrument Air	1.98E-03	1.17E-09	0.03%	99.7%
LOPFA	Loss of Primary Flow - ATWS	8.10E-02	9.28E-10	0.02%	99.8%
LOSPWA	Loss of Offsite Power - Severe Weather Related (ATWS)	2.98E-03	8.18E-10	0.02%	99.8%
MFWLBA	Main Feedwater Line Break - ATWS	2.53E-03	7.67E-10	0.02%	99.8%
MFWLB	Main Feedwater Line Break	2.53E-03	7.58E-10	0.02%	99.8%
SLBC	Steam Line Break In Common Residual Heat Release Line	1.49E-03	6.93E-10	0.02%	99.8%
LOSPPA	Loss of Offsite Power - Plant Centered - ATWS	2.30E-03	6.31E-10	0.02%	99.8%
IAXA	Loss of Station Instrument Air - ATWS	1.98E-03	6.19E-10	0.02%	99.9%
LOSPEA	Loss of Offsite Power - Extreme Weather Related (ATWS)	2.24E-03	6.15E-10	0.02%	99.9%
CRFL	Flood in Control Building Heating Ventilation and Air Conditioning (HVAC) Room	3.29E-06	5.31E-10	0.01%	99.9%
SLOCI	Small LOCA, Isolable	6.98E-04	5.27E-10	0.01%	99.9%
SLBCA	Steam Line Break in Common RHR Line - ATWS	1.49E-03	4.91E-10	0.01%	99.9%
MSV	Main Steam Relief or Safety Valve Opening	9.50E-04	3.69E-10	0.01%	99.9%
ISFLD	Intake Structure Flood in Cubicle D	1.13E-03	3.22E-10	0.01%	99.9%
MSVA	Main Steam Relief or Safety Valve Opening - ATWS	9.50E-04	3.14E-10	0.01%	99.9%
ISFLB	Intake Structure Flood in Cubicle B	6.77E-04	2.85E-10	0.01%	100.0%
SLBIA	Steam Line Break Inside Containment - ATWS	8.31E-04	2.79E-10	0.01%	100.0%
ISFLC	Intake Structure Flood in Cubicle C	6.77E-04	2.79E-10	0.01%	100.0%
IRXA	Loss of Vital Bus I (Red) - ATWS	5.68E-03	2.02E-10	0.01%	100.0%
LB1DA	Loss of Normal 4KV Bus 1D - ATWS	3.51E-03	2.00E-10	0.01%	100.0%
IWXA	Loss of Vital Bus II (White) - ATWS	5.68E-03	1.93E-10	0.005%	100.0%
LB1AA	Loss of Normal 4KV Bus 1A - ATWS	3.51E-03	1.66E-10	0.004%	100.0%
AMSIVA	Closure of All MSIV's - ATWS	1.33E-02	1.50E-10	0.004%	100.0%
ICXA	Loss of Containment Instrument Air - ATWS	1.05E-02	1.19E-10	0.003%	100.0%

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
BVX	Loss of Emergency Switchgear Ventilation	5.82E-09	9.61E-11	0.002%	100.0%
PABF5A	PAB Flood at EI 722 Train A - Not Isolated	1.69E-06	7.69E-11	0.002%	100.0%
PABF5B	PAB Flood at EI 722 Train B - Not Isolated	1.69E-06	7.20E-11	0.002%	100.0%
CVFL	West Cable Vault Flood	1.50E-04	4.16E-11	0.001%	100.0%
CCXA	Loss of Reactor Plant Component Cooling Water - ATWS	2.46E-03	2.72E-11	0.001%	100.0%
WCXA	Loss of River Water Headers A & B - ATWS	1.31E-06	4.39E-12	0.0001%	100.0%
BVXA	Loss of Emergency Switchgear Ventilation - ATWS	5.82E-09	0.00E+00	0.0000%	100.0%

Unit 2:

The Unit 2 CDF breakdown of the remaining 87 internal initiating events (including internal floods) that were not provided in ER, Attachment C-2, page C.2-10, Table 3.1.1.1-1, "BV2REV4 Dominant Initiating Event Contribution to Internal Core Damage," are presented in Table 1.A-2, shown below.

**Table 1.A-2**

**BV2REV4 Remaining Initiating Event Contribution to Internal Core Damage Frequency**

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
PLMFWA	Partial Loss of Main Feedwater - ATWS	2.44E-01	6.18E-08	0.6%	91.1%
LOSPY	Loss of Offsite Power Switchyard Centered - ATWS	7.71E-03	4.64E-08	0.5%	91.5%
EXFW	Excessive Feedwater Flow	8.77E-02	4.39E-08	0.5%	92.0%
IMSIV	Closure of One MSIV	4.22E-02	3.84E-08	0.4%	92.4%
SGFL1B	North Safeguards Train B Area Flood, Isolated	3.65E-04	3.83E-08	0.4%	92.8%
ISI	Inadvertent Safety Injection Initiation	3.79E-02	3.54E-08	0.4%	93.2%
LB2D	Loss of Normal 4160V Bus 2D	5.49E-03	3.31E-08	0.3%	93.5%
LB2A	Loss of Normal 4160V Bus 2A	5.35E-03	3.04E-08	0.3%	93.9%
MLOCAA	Medium Loss of Coolant Accident in Loop A	2.03E-05	2.85E-08	0.3%	94.2%
MLOCAB	Medium Loss of Coolant Accident in Loop B	2.03E-05	2.85E-08	0.3%	94.5%
MLOCAC	Medium Loss of Coolant Accident in Loop C	2.03E-05	2.85E-08	0.3%	94.8%
TLMFW	Total Loss of Main Feedwater	5.12E-02	2.75E-08	0.3%	95.0%
ABFL1B	Auxiliary Building Flood, Service Water Header B Isolated	6.77E-04	2.35E-08	0.2%	95.3%
VPFLB	Service Water Valve Pit Flood, Header B	6.77E-04	2.35E-08	0.2%	95.5%
IRX	Loss of Vital Bus I (Red)	5.01E-03	2.31E-08	0.2%	95.8%
ABFL1A	Auxiliary Building Flood, Service Water Header A Isolated	6.77E-04	2.30E-08	0.2%	96.0%
VPFLA	Service Water Valve Pit Flood, Header A	6.77E-04	2.30E-08	0.2%	96.3%
EXFWA	Excessive Feedwater Flow - ATWS	8.77E-02	2.22E-08	0.2%	96.5%
CCX	Loss of Primary Component Cooling Water	6.27E-03	2.16E-08	0.2%	96.7%
SLOCN	Small LOCA, Non-isolable	7.39E-04	2.09E-08	0.2%	96.9%
IBX	Loss of Vital Bus III (Blue)	5.01E-03	2.09E-08	0.2%	97.2%



Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
IYX	Loss of Vital Bus IV (Yellow)	5.01E-03	2.05E-08	0.2%	97.4%
LPRF	Loss of Primary Flow	4.00E-02	1.96E-08	0.2%	97.6%
D5X	Loss of 125V DC Battery 2-5 Supply	1.03E-02	1.72E-08	0.2%	97.8%
LOSPW	Loss of Offsite Power Severe Weather - ATWS	2.85E-03	1.71E-08	0.2%	97.9%
IWX	Loss of Vital Bus II (White)	5.01E-03	1.71E-08	0.2%	98.1%
CVFLB	Cable Vault Flood from Service Water Header B	6.02E-06	1.60E-08	0.2%	98.3%
CVFLA	Cable Vault Flood from Service Water Header A	6.02E-06	1.58E-08	0.2%	98.4%
LOSPP	Loss of Offsite Power Plant Centered - ATWS	2.30E-03	1.38E-08	0.1%	98.6%
TLMFWA	Total Loss of Main Feedwater - ATWS	5.12E-02	1.30E-08	0.1%	98.7%
IMSIVA	Closure of One MSIV - ATWS	4.22E-02	1.02E-08	0.1%	98.8%
ISIA	Inadvertent Safety Injection - ATWS	3.79E-02	9.11E-09	0.1%	98.9%
DPXA	Loss of Emergency 125V DC Purple	1.03E-02	7.83E-09	0.1%	99.0%
LCV	Loss of Condenser Vacuum	1.36E-02	7.25E-09	0.1%	99.1%
TTRIPA	Turbine/Generator Trip - ATWS	4.49E-01	6.91E-09	0.1%	99.2%
LLOCAA	Large Loss of Coolant Accident in Loop A	2.40E-06	6.05E-09	0.1%	99.2%
LLOCAB	Large Loss of Coolant Accident in Loop B	2.40E-06	6.05E-09	0.1%	99.3%
LLOCAC	Large Loss of Coolant Accident in Loop C	2.40E-06	6.05E-09	0.1%	99.4%
CPEXC	Core Power Excursion	1.20E-02	5.85E-09	0.1%	99.4%
SLBD	Steam Line Break Outside Containment	4.65E-03	4.90E-09	0.1%	99.5%
SLB1	Steam Line Break Inside Containment	8.46E-04	4.56E-09	0.05%	99.5%
TBFL	Turbine Building Flood	7.59E-03	3.83E-09	0.04%	99.6%
IBXA	Loss of Vital Bus III (Blue) - ATWS	5.01E-03	3.70E-09	0.04%	99.6%
LCVA	Loss of Condenser Vacuum - ATWS	1.36E-02	3.43E-09	0.04%	99.6%
CBFL	Control Building Flood	3.61E-07	2.82E-09	0.03%	99.7%
SLOCI	Small LOCA, Isolable	7.39E-04	2.79E-09	0.03%	99.7%
LOSPGA	Loss of Offsite Power Grid Related - ATWS	1.33E-02	2.78E-09	0.03%	99.7%
IYXA	Loss of Vital Bus IV (Yellow) - ATWS	5.01E-03	2.75E-09	0.03%	99.7%
MFWLB	Main Feedwater Line Break	2.66E-03	2.50E-09	0.03%	99.8%
AMSIV	Closure of All MSIVs	4.91E-03	2.20E-09	0.02%	99.8%
SLBC	Steam Line Break in Common RHS Line	1.55E-03	2.15E-09	0.02%	99.8%
LOSPYA	Loss of Offsite Power Switchyard Centered - ATWS	7.71E-03	1.61E-09	0.02%	99.8%

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
ICXA	Loss of Containment Instrument Air Supply - ATWS	8.59E-02	1.32E-09	0.01%	99.8%
SLBDA	Steam Line Break Outside Containment - ATWS	4.65E-03	1.23E-09	0.01%	99.9%
ISFLD	Intake Structure Flood Cube D	1.13E-03	1.22E-09	0.01%	99.9%
MSV	Main Steam Relief or Safety Valve Opening	9.89E-04	9.72E-10	0.01%	99.9%
AOXA	Loss of Emergency 4160V AC Orange - ATWS	1.43E-02	8.53E-10	0.01%	99.9%
ISFLC	Intake Structure Flood Cube C	6.77E-04	8.19E-10	0.01%	99.9%
BPXA	Loss of Emergency 4160V AC Purple - ATWS	1.40E-02	7.46E-10	0.01%	99.9%
IAX	Loss of Station Instrument Air Supply	1.36E-03	6.75E-10	0.01%	99.9%
MFWLBA	Main Feedwater Line Break - ATWS	2.66E-03	6.37E-10	0.01%	99.9%
LPRFA	Loss of Primary Flow - ATWS	4.00E-02	6.12E-10	0.01%	99.9%
LOSPWA	Loss of Offsite Power Severe Weather - ATWS	2.85E-03	5.93E-10	0.01%	99.9%
DOXA	Loss of Emergency 125V DC Orange - ATWS	1.03E-02	5.64E-10	0.01%	99.9%
LOSPPA	Loss of Offsite Power Plant Centered - ATWS	2.30E-03	4.78E-10	0.01%	99.9%
LOSPEA	Loss of Offsite Power Extreme Weather - ATWS	2.24E-03	4.66E-10	0.005%	100.0%
SGTRAA	Loop A Steam Generator Tube Rupture - ATWS	1.61E-03	4.24E-10	0.004%	100.0%
SGTRBA	Loop B Steam Generator Tube Rupture - ATWS	1.61E-03	4.24E-10	0.004%	100.0%
SGTRCA	Loop C Steam Generator Tube Rupture - ATWS	1.61E-03	4.24E-10	0.004%	100.0%
SLBCA	Steam Line Break in Common RHS Line - ATWS	1.55E-03	4.10E-10	0.004%	100.0%
ISFLB	Intake Structure Flood Cube B	6.77E-04	4.04E-10	0.004%	100.0%
IAXA	Loss of Station Instrument Air Supply - ATWS	1.36E-03	3.43E-10	0.004%	100.0%
MSVA	Main Steam Relief/Safety Valve Opens - ATWS	9.89E-04	2.61E-10	0.003%	100.0%
D5XA	Loss of 125V DC Battery 2-5 Supply - ATWS	1.03E-02	2.49E-10	0.003%	100.0%
IRXA	Loss of Vital Bus I (Red) - ATWS	5.01E-03	2.31E-10	0.002%	100.0%
SLB1A	Steam Line Break Inside Containment - ATWS	8.46E-04	2.27E-10	0.002%	100.0%
IWXA	Loss of Vital Bus II (White) - ATWS	5.01E-03	2.21E-10	0.002%	100.0%
CPEXCA	Core Power Excursion - ATWS	1.20E-02	1.83E-10	0.002%	100.0%
WBXA	Loss of Service Water Train B - ATWS	4.72E-03	1.09E-10	0.001%	100.0%

Initiator	Description	Initiating Event Frequency	Contribution to Internal CDF	Percent of Internal CDF	Cumulative Percent of Internal CDF
LB2DA	Loss of Normal 4160V Bus 2D - ATWS	5.49E-03	1.04E-10	0.001%	100.0%
LB2AA	Loss of Normal 4160V Bus 2A - ATWS	5.35E-03	9.61E-11	0.001%	100.0%
WAXA	Loss of Service Water Train A - ATWS	4.15E-03	9.55E-11	0.001%	100.0%
CCXA	Loss of Primary Component Cooling Water - ATWS	6.27E-03	9.48E-11	0.001%	100.0%
ABFL2B	Auxiliary Building Flood From Service Water Header B, Non-isolated	2.20E-06	7.55E-11	0.001%	100.0%
ABFL2A	Auxiliary Building Flood From Service Water Header A, Non-isolated	2.20E-06	7.53E-11	0.001%	100.0%
AMSIVA	Closure of All MSIVs - ATWS	4.91E-03	7.41E-11	0.001%	100.0%
WCXA	Loss of Both Service Water Trains A & B - ATWS	2.61E-06	7.64E-12	0.0001%	100.0%

RESPONSE SAMA-1.b

**b. Clarify whether anticipated transient without scram (ATWS) events are modeled in the external event analysis. Provide the ATWS and station blackout (SBO) CDF for both internal and external event initiators.**

The BVPS PRA models do not include anticipated transient without scram (ATWS) events in their external event analyses. The decision to not include the ATWS events was based on the fairly low initiating event frequencies of external events (typically in the 1E-04 to 1E-08 range) along with the low failure probabilities of the reactor trip system (1E-06 when offsite power is lost; 3E-06 when all support is available; and, 1E-03 with the loss of a single solid-state protection system (SSPS) train). The ATWS and station blackout (SBO) CDF contributions for internal and external events are provided in Tables 1.B-1 and 1.B-2, shown below.

**Table 1.B-1**

Unit 1 ATWS and SBO CDF Contribution

	Contribution to Internal CDF (/year)	Contribution to External CDF (/year)	Contribution to Total CDF (/year)
ATWS	3.85E-07	-	3.85E-07
SBO	2.62E-07	7.41E-06	7.67E-06

**Table 1.B-2**

Unit 2 ATWS and SBO CDF Contribution

	Contribution to Internal CDF (/year)	Contribution to External CDF (/year)	Contribution to Total CDF (/year)
ATWS	1.57E-07	-	1.57E-07
SBO	8.14E-07	4.54E-06	5.35E-06

RESPONSE SAMA-1.c

- c. Provide a discussion on the loss of containment instrument air. Identify any differences in plant features or PRA models/assumptions that cause the loss of containment instrument air initiator (ICX) to be a larger contributor in Unit 2 than Unit 1.

A breakdown of the loss of containment instrument air core damage frequency for both BVPS Units is provided in Table 1.C-1, shown below:

**Table 1.C-1**  
 BVPS Breakdown of Loss of Containment Instrument Air (ICX)

BVPS	Initiating Event Frequency (per yr.)	Core Damage Frequency (per yr.)	Conditional Core Damage Probability	Percent Contribution to Internal CDF	Percent Contribution to Total CDF
Unit 1	1.05E-02	6.19E-09	5.88E-07	0.2%	0.03%
Unit 2	8.59E-02	2.94E-07	3.42E-06	3.1%	1.2%

The table shows that the Unit 2 initiating event frequency is approximately a factor of 8 times that of the Unit 1 frequency. This difference is accounted for by physical plant differences and PRA modeling differences.

The BVPS physical plant differences contributing to the loss of containment instrument air initiating event frequencies consist of the following:

- There are no Unit 1 containment instrument air compressors; air is supplied through a normally-opened cross-tie to the station instrument air system.
- There is one Unit 1 containment air receiver, which has an associated relief valve, located inside the containment building.
- There are two Unit 2 containment instrument air compressors located outside the containment building.
- There are two Unit 2 containment air receivers, each of which has an associated relief valve; one receiver is located inside the containment building, and the other receiver is located outside the containment building.

Based on the above physical plant differences, the BVPS PRA modeling differences for the loss of containment instrument air initiating event frequencies consist of the following:

- Since Unit 1 has only one containment air receiver and associated relief valve, it has only one flow-diversion path—through the premature opening of the relief valve. A review of the initiating event cutsets shows that the initiating event frequency is dominated by the premature opening of this relief valve (failure probability of  $9.49\text{E-}03$  based on 1-year mission time), which has a 90.4% contribution to the initiating event frequency.
- In contrast, Unit 2 has two containment air receivers and associated relief valves, so it has two flow-diversion paths through the premature opening of each relief valve. A review of the initiating event cutsets shows that the initiating event frequency is also dominated by the premature opening of these relief valves (each having a failure probability of  $4.18\text{E-}02$  based on 1-year mission time), which have a 97.4% contribution to the initiating event frequency.

Other BVPS PRA modeling differences contributing to the dissimilarities in the relief valve premature opening failure probabilities consist of the following:

- The Unit 1 failure frequency for the premature opening of a relief valve is  $1.17\text{E-}06$  per hour, which is based on a two-stage Bayesian update of the generic prior failure rate with actual plant failure data.
- The Unit 2 failure frequency for the premature opening of a relief valve is  $6.06\text{E-}06$  per hour, which is the generic prior failure rate since no failures were identified during reviews of the plant failure data.
- Performing a two-stage Bayesian update of the generic prior failure rate with zero failures in an estimated 1.18 million relief valve operating hours during the current PRA model data update period (27 valves operating from January 1, 2001, through December 31, 2005) would have produced a relief valve premature opening failure rate of  $1.35\text{E-}06$  per hour at Unit 2, which is more comparable to the Unit 1 value.
- There are also some differences in the availability factors used in evaluating the initiating event frequencies between the two Units. In comparing the initiating event frequencies, these differences were factored out by assuming 100% availability for each unit.
- By incorporating the Bayesian updated relief valve premature opening failure rate of  $1.35\text{E-}06$  per hour and assumed 100% availability factor at Unit 2, the revised initiating event frequency for the loss of containment instrument air would be about  $2.61\text{E-}02$  per year. This would result in an approximate factor of 2 times

that of the Unit 1 frequency ( $1.14\text{E-}02$  per year assuming 100% availability), which is expected due to twice the number of air receiver relief valves, compared to the factor of 8 for the BV2REV4 base model.

Table 1.C-1 also shows that the Unit 2 loss of containment instrument air conditional core damage probability (CCDP) is approximately a factor of 6 times that of the Unit 1 CCDP. Reviews of the ICX top dominant sequences at both Units provided some insights into why these CCDPs are different, as well as how they are attributed to both physical plant differences and PRA modeling differences.

- The top dominant ICX sequence at Unit 1 has a core damage frequency of  $2.03\text{E-}09$  per year, and a 33% contribution to the ICX CDF. It is initiated by a loss of containment instrument air that results in the closure of the reactor coolant pump (RCP) thermal barrier cooling valves and letdown isolation valves, followed by the probabilistic failure of river water train A ( $7.10\text{E-}06$ ) and the conditional failure of river water train B ( $1.50\text{E-}01$ ). This scenario leads to the failure of the pumps that support RCP thermal barrier cooling and seal injection, resulting in a 182 gallon per minute (gpm) per RCP seal loss of coolant accident (LOCA) ( $1.98\text{E-}01$  probability), which eventually leads to core damage without any reactor coolant system (RCS) makeup available by the safety injection pumps.
- The similar sequence at Unit 2 is ranked 57<sup>th</sup> amongst the ICX sequences, and has a core damage frequency of  $1.10\text{E-}10$  per year with a 0.04% contribution to the ICX CDF. The accident sequence progression is the same as that for Unit 1; however, the probabilistic failure of service water train A is  $1.15\text{E-}05$  and the conditional failure of service water train B is  $6.16\text{E-}04$ . These values are attributed to both physical plant differences and PRA modeling differences.
- At Unit 1 there is normally only one river water pump in service providing flow to both headers through a common cross-tie pipe. Additionally, there is a common discharge flow path for the headers containing a single isolation valve. As such, these single failures are the dominant contributors to both of the river water headers failing in the Unit 1 PRA model, so the conditional failure probability of the B header failing given the failure of the A header is fairly high ( $1.50\text{E-}01$ ).
- Unit 2 normally has a service water pump operating on each header and does not have any single component failures common to both headers, so the conditional failure probability of the B header is much lower ( $6.16\text{E-}04$ ).
- In addition, the Unit 1 failure frequency for pipe ruptures with greater than a 3-inch diameter is  $1.53\text{E-}08$  per hour, which is based on a one-stage Bayesian update of the generic prior failure rate of  $8.60\text{E-}10$  failures per hour with 1 actual failure in an estimated 6.52 million operating hours (from January 1, 1980, through December 31, 2005).

- The Unit 2 failure frequency for pipe ruptures in piping of greater than 3-inches in diameter is  $7.79E-10$  per hour, which is based on a one-stage Bayesian update of the same generic prior failure rate as Unit 1 with zero actual failures in an estimated 2.65 million operating hours (from November 17, 1987, through December 31, 2005).
- The top dominant ICX sequence at Unit 2 has a core damage frequency of  $1.85E-07$  per year, and a 63% contribution to the ICX CDF. The sequence is initiated by a loss of containment instrument air that results in the closure of the RCP thermal barrier cooling valves and letdown isolation valves, thereby swapping over the charging/high-head safety injection (HHSI) pump suction source to the refueling water storage tank (RWST) on low volume control tank (VCT) level. These events are then followed by the probabilistic failure of the RWST suction supply to the high head and low head safety injection (LHSI) pumps ( $1.18E-05$ ), which leads to the failure of the charging/HHSI pumps and loss of RCP seal cooling. Due to the loss of all RCP seal cooling, a 182 gpm per RCP seal LOCA occurs ( $1.98E-01$  probability) and eventually leads to core damage, without any RCS makeup available by the safety injection pumps.
- The similar sequence at Unit 1 is the third highest-ranked ICX sequence, and has a core damage frequency of  $8.60E-10$  per year with a 14% contribution to the ICX CDF. The accident sequence progression is the same as that for Unit 2; however, the probabilistic failure of the RWST suction supply to the high head and low head safety injection pumps is only  $4.38E-07$ . This value is attributed to both physical plant differences and PRA modeling differences.
- At Unit 2, the RWST has a manual isolation valve in the common suction supply line to the high head and low head safety injection pumps, and is, therefore, modeled in the RWST top event fault tree. The inadvertent closure of this isolation valve has a failure probability of  $1.13E-05$ , which contributes to 95% of the RWST failure probability, with the remainder attributed to the tank rupture probability ( $4.85E-07$ ).
- Unit 1 does not have a common suction supply line manual isolation valve arrangement, but rather the HHSI and LHSI suction lines have separate manual isolation valves. Therefore, the RWST failure probability is only due to the tank rupture probability ( $4.38E-07$ ).



RESPONSE SAMA-1.d

- d. Significant CDF reduction was achieved as a result of Revision 3 update for Unit 1, i.e., a reduction in internal event CDF from 6.24E-5 per year in Revision 2 to 7.45E-6 per year in Revision 3. Discuss the reasons for the reduction in CDF and the extent to which these changes were considered in the Westinghouse Owner's Group peer review.**

The major Level 1 changes incorporated into the Beaver Valley Unit 1 Revision 3 PRA model are discussed in the ER, Attachment C-1, Section 3.1.1.2 of the Unit 1 SAMA Analysis. The individual effect on the total CDF by incorporating each of the changes has not been analyzed. However, each change is listed in order of expected importance in the reduction of the CDF, with the first bulleted item, the modified WCAP-15603 RCP seal LOCA model, being the most important. It should also be noted that the last bulleted item, the 2.5% of maintenance unavailability for the emergency diesel generators, would be expected to cause an increase in the CDF. The Revision 3 PRA model also incorporated the Westinghouse Owner's Group (WOG) PRA peer review resolutions to the Category A and B Facts and Observations (F&Os). In particular, the 2nd, 6th, and 10th bulleted items were changes made to the PRA model as a direct result of incorporating the WOG PRA peer review resolutions to some F&Os. In addition, the 1st, 5th, and 7th bulleted items were changes made to the PRA model as a result of a self assessment of the PRA models in preparation of the WOG PRA peer review. In the 4th bulleted item, the data sources were changes made as a result of a self assessment, while the update period was based on the WOG PRA peer review.

Table 1.D-1 (shown below) provides an assessment for some of the Revision 3 PRA model changes (bulleted items) and their impact on the total reduction in the core damage frequency for the significant accident categories. As expected, the first two bulleted items, which dealt with the changes made to the RCP seal LOCA model, helped to contribute to nearly a 60-percent reduction of the BV1REV2 total CDF. The third bulleted item (removal of the charging pump ventilation dependency) also helped in reducing the total CDF by nearly 30 percent of the BV1REV2 total. The fourth bulleted item does not appear on the table (changes to initiating events data based on WCAP-15210); however, it resulted in major reductions in some initiating event frequencies; but also increased some initiating event frequencies, too. The 8th and 9th bulleted items also do not appear on the table (motor operated valve (MOV) failure rates and SSPS failure probabilities) and are far reaching model changes, which virtually impact all core damage sequences due to MOV actuations and SSPS signals required to mitigate an accident. It should be pointed out that the SSPS model changes were made based on License Amendment Number 141 for the Unit 2 Slave Relay Surveillance Test Interval Extension, dated May 14, 2004 (Agencywide Documents Access and Management System (ADAMS) accession number ML041030082).

**Table 1.D-1**

Impact of BVPS-1 PRA Model Changes from Revision 2 to Revision 3

Accident Category	BV1REV2 PRA Model		BV1REV3 PRA Model		Delta CDF in PRA Models	Percent Reduction of BV1REV2 Total CDF	Numbered Order of Bulleted Item(s) Covered in SAMA
	Core Damage Frequency	Percent Contribution to Total CDF	Core Damage Frequency	Percent Contribution to Total CDF			
Total CDF (Internal & External Events)	8.50E-05	-	2.34E-05	-	-6.2E-05	72.5%	
RCP Seal LOCA	6.43E-05	75.6%	1.51E-05	64.5%	-4.9E-05	57.9%	1st, 2nd
Loss of HHSI Pump HVAC	2.41E-05	28.4%	N/A*	N/A*	-2.4E-05	28.4%	3rd
Station Blackout	1.08E-05	12.7%	6.04E-06	25.8%	-4.8E-06	5.6%	2nd, 5th
ATWS	2.74E-06	3.2%	1.88E-07	0.8%	-2.6E-06	3.0%	6th, 7th
ISLOCA	3.76E-07	0.4%	7.97E-08	0.3%	-3.0E-07	0.3%	10th
* The HHSI pump HVAC support dependency was removed from the BV1REV3 PRA model.							

RESPONSE SAMA-1.e

- e. It is indicated that the heat up rates for the switchgear rooms is slower than what was assumed during the Individual Plant Examination (IPE). The heatup times are now about 5 hours in Unit 1 (per discussion of SAMA 181 in Table 6-1) and more than 24 hours in Unit 2 (per discussion in Section 3.1.1.2). Describe the plant features or modeling that causes the difference in heat up rates between the units.

The calculations used for the emergency switchgear room heat up rates were performed using the principles of conservation of energy and heat transfer to calculate the exiting room air temperature over time. The heat up rates calculated in the Individual Plant Examination (IPE) were based on conservative design heat loads, while the current PRA models are based on empirical heat loads derived from actual temperature readings obtained during temporary operating procedure ventilation tests. The current analysis results for Unit 1 show that the emergency switchgear room would heat up to 120 °F in 6.8 hours following a loss of emergency switchgear room ventilation. The current analysis results for Unit 2 show that the emergency switchgear room would heat up to 118.5 °F in 24 hours following a loss of emergency switchgear room ventilation. The major differences in the room heat up rates between the two Units are due to the following plant features and modeling provided in Table 1.E-1, shown below.

**Table 1.E-1**  
Emergency Switchgear Room Heat Up Rate Input Comparison

Plant / Modeling Differences	Unit 1	Unit 2
Empirical heat loads (BTU/hr)	290,000	345,000
Assumed heat sinks (lbs)	200,000	150,000
Estimated room volumes (ft <sup>3</sup> )	57,852	85,310
Assumed initial room temperature (°F)	80	90
Assumed outside air temperature (°F)	90	79.1 to 99.1
Credited Battery Room HVAC flow rate (cfm)	0	4,800

As shown in Table 1.E-1, Unit 1 has a lower heat load, more assumed heat sinks, and a lower assumed initial room temperature, which should tend to increase the time to reach 120°F following a loss of emergency switchgear ventilation when compared to Unit 2. However, the Unit 1 room volume for the emergency switchgear room is approximately 68 percent of the Unit 2 room volume, which would result in the Unit 1 room heating faster. A modeling feature that leads to the differences in the room heat up rates is that

a constant 90 °F outside air temperature was assumed for the Unit 1 analysis, while a diurnal outside air temperature fluctuating between 79.1 °F and 99.1 °F during a 24-hour period was assumed for the Unit 2 analysis. A difference in the plant features between the two Units is that Unit 2 has redundant battery room exhaust fans that are separate from the emergency switchgear fans, unlike Unit 1, which uses the same fans to exhaust the emergency switchgear and battery rooms. Crediting these Unit 2 battery room ventilation fans for removing heat from the emergency switchgear room aids in reducing the room heat-up rates compared to Unit 1, which assumed no flow following the loss of the emergency switchgear ventilation.

#### RESPONSE SAMA-1.f

- f. Discuss the peer reviews performed (internal and/or external to FENOC) and quality controls applied to the external event models, to the Level 2 PRA models, and to the internal event PRA revisions subsequent to the Westinghouse Owner's Group and Nuclear Energy Institute peer reviews (i.e., Revisions 3 and 4 for Unit 1 and Revisions 3B and 4 for Unit 2.)**

The quality controls applied to both of the BVPS PRA models (which integrate internal, external, Level 1, and Level 2) are governed by BVPS site procedures. These procedures provide direction for maintaining and updating the PRA models to ensure that they represent current plant design and operation. The BVPS PRA models are developed and reviewed in accordance with site procedures. In addition, the RISKMAN software used to build and quantify the PRA models, is certified under the vendor's 10 CFR 50 Appendix B software quality assurance program, and is classified as FENOC Category B software, which requires that the installation be verified and validated, and that configuration control be maintained.

Reviews were performed on the external event models, Level 2 PRA models, and internal event model revisions subsequent to the Westinghouse Owner's Group (WOG) peer review conducted in July 2002, using Nuclear Energy Institute (NEI) guidance. These included reviews performed during the NRC Significance Determination Process (SDP) Phase 2 Notebook Benchmarking Visit in July 2003, an NRC Extended Power Uprate (EPU) PRA models audit conducted in October 2005, a Regulatory Guide (RG) 1.200 gap-assessment performed in October 2007, and a Human Reliability Analysis (HRA) focused peer review performed in October 2007. These reviews are briefly described below.

- NRC SDP Phase 2 Notebook Benchmarking Visit

During July 2003, NRC staff and contractors visited BVPS to compare the Unit 1 and Unit 2 SDP Phase 2 notebooks and BVPS PRA model results to ensure that the SDP notebooks were generally conservative. Since the BVPS PRA models (Revision 3 for Unit 1 - BV1REV3, and Revision 3B for Unit 2 - BV2REV3B)

included most external initiating events, sensitivity studies were performed to assess the impact of these initiators on SDP color determinations. In addition, the results from analyses using the NRC's draft Revision 3i Standardized Plant Analysis Risk (SPAR) models for BVPS Unit 1 and Unit 2 were compared with the BVPS risk models.

The benchmarking visit identified that there was a strong correlation between the Phase 2 SDP Notebooks and the BVPS PRA models. The results (see report references, listed below) indicate that the BVPS Unit 1 and Unit 2 Phase 2 notebooks were generally more conservative in comparison to the BVPS PRA models. The Unit 1 revision 1 SDP notebook will capture 90% (results matched or overestimated the BVPS PRA by one order of magnitude) of the risk significance of inspection findings, while the Unit 2 revision 1 SDP notebook will capture 96%. SDP Report references include:

1. U.S. Nuclear Regulatory Commission Report, "Results of the Beaver Valley Power Station Unit 1 SDP Phase 2 Notebook Benchmarking Visit," September 24, 2003 (ADAMS accession number ML032680999).
2. U.S. Nuclear Regulatory Commission Report, "Results of the Beaver Valley Power Station Unit 2 SDP Phase 2 Notebook Benchmarking Visit," September 24, 2003 (ADAMS accession number ML032681044).

- NRC EPU PRA Models Audit

As part of the review of the BVPS Unit 1 and Unit 2 Extended Power Uprate License Amendment Request, the NRC staff performed an audit to assess the BVPS PRA models (BV1REV3 and BV2REV3B) in October 2005. One focus of this audit was PRA quality, particularly with respect to maintaining configuration control of the models.

The results of the NRC staff review were documented in Section 2.13 of the NRC Safety Evaluation Report for the Extended Power Uprate License Amendments 275/156 for BVPS Unit 1 and Unit 2. The review determined that, "the NRC staff finds that the PRA used in support of the EPU is of sufficient quality, scope, and level of detail to analyze the risks stemming from the EPU, consistent with the guidance in RG 1.174 (Section 2.2.3), SRP Chapter 19 (Sections III.2.2.2, III.2.2.3, III.2.2.4, and Appendix A) and SRP Chapter 19.1, and is, therefore, acceptable."

- Regulatory Guide 1.200 Gap Assessment

A RG 1.200 gap assessment of the BVPS Unit 2 PRA model was conducted by FENOC and Westinghouse personnel in October 2007. The primary objective of this review was to provide a baseline assessment of how well the Revision 4 BVPS Unit 2 PRA model (BV2REV4) and documentation meet the supporting

requirements in Addendum B to the ASME PRA Standard, and to determine the applicable Capability Category (CC) for each of the Supporting Requirements (SRs). To the extent that the PRA modeling methodologies are equivalent, this self-assessment is also applicable to the Unit 1 PRA model (BV1REV4).

The assessment was conducted using the guidance in Appendices B and D of NEI 00-02, "Industry PRA Peer Review Process Guidance," which is consistent with Appendix B of NRC RG 1.200 Revision 1, considering regulatory interpretations of the ASME PRA Standard as noted in Appendix A of RG 1.200.

The assessment was performed using an established multi-step process:

1. The first step of this review was to determine if the SRs in the PRA Standard were adequately addressed in the original peer review. This was completed by identifying if the SRs could be mapped to an element of the original peer review using NEI 00-02, and whether the Regulatory Guide 1.200 Appendix A or Appendix B provided any clarifications requiring additional review of mapped elements.
2. If the SR mapped to an element of the original peer review and no RG 1.200 clarifications exist, then the Reviewer's Notes from the original peer review and any F&Os for that element were reviewed. In the case where F&Os were written from the original peer review, the BVPS-2 F&O resolution was reviewed to determine if it was adequate to satisfy the requirements of the ASME PRA Standard.
3. If the SR mapped to an element of the original peer review and RG 1.200 clarifications exist, then the Reviewer's Notes from the original peer review and any F&Os for that element were reviewed along with the PRA documentation to determine if the RG 1.200 clarifications were satisfied. Again, in the case where F&Os were written from the original peer review, the BVPS-2 F&O resolution was reviewed to determine if it was adequate to satisfy the requirements of the ASME PRA Standard.
4. If the SR did not map to an element of the original peer review, then the PRA documentation was reviewed to independently determine whether the SR from the ASME PRA Standard and any RG 1.200 clarifications were satisfied. These detailed reviews were based on several samples of the PRA model and were not a comprehensive review of the entire model.
5. When issues that identified deficiencies with respect to meeting Capability Category II of the PRA Standard or suggestions for improvements were identified, new F&Os were developed to identify the deficiency, categorize the significance level, and to suggest a path for resolution.

- Human Reliability Analysis (HRA) Focused Peer Review

The results of the WOG peer review identified a number of issues, but concluded that the BVPS PRA was sufficient to use for risk-informed applications as long as it was supported by additional application-specific deterministic evaluations and the identified issues were addressed. One of the key areas of concern identified in the WOG peer review was that the BVPS HRA had several key deficiencies associated with the Success Likelihood Index Methodology (SLIM). To address these deficiencies, BVPS changed the HRA methodology from SLIM to the EPRI HRA Calculator, and, in doing so, required that a follow-on peer review be performed in accordance with the ASME PRA Standard. Therefore, in October 2007, a focused peer review of the BVPS HRA to determine compliance with Addendum B to the ASME PRA Standard and RG 1.200, Revision 1, was performed using NEI 05-04. The scope of this review was to assess the BVPS HRA performed for Revision 4 of the BVPS PRAs (BV1REV4 and BV2REV4) against the Human Reliability element of the ASME PRA Standard, which contains a total of thirty-five SRs under nine High Level Requirements (HLRs).

New Findings (i.e., F&Os) were prepared to document any new issues that were identified during the review of the BVPS HRA against the ASME PRA Standard.

RESPONSE SAMA-1.g

**g. Identify the major shared systems and components between the two units. Discuss how their risk contributions are accounted for in the PRA.**

The major shared systems and components between the two Beaver Valley Units that are modeled in the BVPS PRAs include the Emergency Response Facility (ERF) substation, and the emergency diesel generators (EDGs) through a common cross-tie cable during station blackout (SBO) events.

The ERF substation risk contribution is accounted for in each PRA model through the use of separate top events. This top event is designated as BK in each of the PRA models, and basically uses the same ERF substation model. The loss of this system by itself does not lead to a plant trip at either Unit, so the risk contribution is accounted for separately in each PRA model. Each Unit's risk achievement worth (RAW) and risk reduction worth (RRW) for the ERF substation top event BK is provided in Table 1.G-1, shown below.

**Table 1.G-1**  
Risk Importance Measures for the ERF Substation

BVPS	Risk Achievement Worth	Risk Reduction Worth *
Unit 1	1.331E+00	1.003E+00
Unit 2	1.330E+00	1.001E+00
* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship: $RRW = 1 / (1 - FV)$		

The EDGs are components that can be shared between the Units whenever there is an SBO event, through a common cross-tie cable. This cross-tie allows any single available emergency diesel generator at either Unit to supply 4 kV power to vital shutdown equipment at both Units during a postulated total loss of offsite power to both Beaver Valley Units, coincident with the loss of both EDGs at one Unit and one EDG at the other Unit. This electric power cross-tie capability is designated as top event XT in each of the PRA models, and is only queried in the PRA model event trees when both of the emergency AC power trains have failed.

When evaluating the emergency diesel generator cross-tie between the two Units, it is assumed that if a loss of offsite power has occurred at one Unit, resulting in an SBO condition, the opposite Unit would also have a guaranteed loss of offsite power (i.e., dual-Unit loss of offsite power event) and subsequent start of the EDGs. That is to say, failure probabilities regarding the supply of AC power from the opposite Unit's emergency diesel generator are based on the conditional probabilities given that a loss of offsite power has occurred.

Therefore, there are some dependencies in the risk contributions for top event XT through the use of the opposite Unit's AC power supply from the EDGs. These dependencies are modeled as basic events in the PRA models. At Unit 1, the basic event XXBV2DG is used to model when power from the Unit 2 EDGs is not available. At Unit 2, the basic event XXBV1DG is used to model when power from the Unit 1 EDGs is not available. Each Unit's RAW and RRW for the opposite Unit's EDG power supply basic event is provided in Table 1.G-2, shown below.



**Table 1.G-2**  
 Risk Importance Measures for the EDG Cross-tie Basic Events

<b>BVPS</b>	<b>Basic Event (BE)</b>	<b>BE Risk Achievement Worth</b>	<b>BE Risk Reduction Worth *</b>
Unit 1	XXBV2DG	1.224E+00	1.001E+00
Unit 2	XXBV1DG	1.239E+00	1.002E+00
* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship: $RRW = 1 / (1 - FV)$			

RESPONSE SAMA-1.h

**h. Provide the CDF for internal floods and provide a breakdown and summary of the top flood scenarios.**

Unit 1

The CDF for all internal floods is 1.24E-07 per year at Unit 1. A core damage frequency breakdown as a percentage of floods and total CDF for all of the Unit 1 internal floods is provided in Table 1.H-1, shown below.

**Table 1.H-1**  
Unit 1 Internal Flood Initiating Events CDF Breakdown

Rank	Flood Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Floods CDF	Percentage of Total CDF	Description
1	PABF2A	2.54E-05	2.02E-08	16.4%	0.10%	PAB Flood At EI 735 Train A - Not Isolated
2	PABF2B	2.54E-05	2.01E-08	16.2%	0.10%	PAB Flood At EI 735 Train B - Not Isolated
3	PABF3A	3.89E-04	1.79E-08	14.5%	0.09%	PAB Flood At EI 722 Train A - Early Isolation
4	PABF1A	3.84E-04	1.78E-08	14.4%	0.09%	PAB Flood At EI 735 River Water Train A - Isolated
5	PABF1B	3.84E-04	1.67E-08	13.5%	0.09%	PAB Flood At EI 735 River Water Train B - Isolated
6	PABF3B	3.89E-04	1.56E-08	12.6%	0.08%	PAB Flood At EI 722 Train B - Early Isolation
7	TBFL	7.71E-03	9.27E-09	7.50%	0.05%	Turbine Building Flood
8	PABF4A	3.32E-05	1.54E-09	1.25%	0.01%	PAB Flood At EI 722 Train A - Late Isolation
9	PABF4B	3.32E-05	1.44E-09	1.16%	0.01%	PAB Flood At EI 722 Train B - Late Isolation
10	ISFLA	9.01E-04	1.39E-09	1.12%	0.01%	Intake Structure Flood In Cubicle A
11	CRFL	3.29E-06	5.32E-10	0.43%	0.00%	Flood In Control Bldg HVAC Room
12	ISFLD	1.13E-03	3.27E-10	0.26%	0.00%	Intake Structure Flood In Cubicle D
13	ISFLB	6.77E-04	2.90E-10	0.23%	0.00%	Intake Structure Flood In Cubicle B
14	ISFLC	6.77E-04	2.83E-10	0.23%	0.00%	Intake Structure Flood In Cubicle C
15	PABF5A	1.69E-06	7.85E-11	0.06%	0.00%	PAB Flood At EI 722 Train A - Not Isolated
16	PABF5B	1.69E-06	7.35E-11	0.06%	0.00%	PAB Flood At EI 722 Train B - Not Isolated
17	CVFL	1.50E-04	4.35E-11	0.04%	0.00%	West Cable Vault Flood
	<b>ALL FLOODS</b>		<b>1.24E-07</b>	<b>100%</b>	<b>0.63%</b>	

As shown in Table 1.H-1, the top six floods at Unit 1 are all associated with floods in the primary auxiliary building (PAB). Together these top six floods contribute to almost 88% of the Unit 1 flood core damage frequency, but less than 0.6% of the total core damage frequency. As such, they were not considered to be significant enough to warrant any further consideration for SAMA evaluations.

A summary of the top six flood initiating event scenarios for Unit 1 is provided in Table 1.H-2, shown below, and the discussion that follows.

**Table 1.H-2**  
 Unit 1 Top Six Internal Flood Scenarios

<b>Flood</b>	<b>Description</b>	<b>Cause of Plant Trip</b>	<b>Plant Model Impact</b>
PABF1A	PAB EI 735 River Water Header A Break	Reactor Trip	Loss of River Water (RW) header A, Charging Pump B failed, and RWST supply to charging failed
PABF1B	PAB EI 735 River Water Header B Break	Reactor Trip	Loss of RW header B, Charging Pump B failed, and RWST supply to charging failed
PABF2A	PABF1A for 15-30 min	Reactor Trip	Same as PABF1A plus all Reactor Plant Component Cooling Water (CCR) pumps failed
PABF2B	PABF1B for 15-30 min	Reactor Trip	Same as PABF1B plus all CCR pumps failed
PABF3A	PAB EI 722 River Water Header A Break	Reactor Trip	Loss of RW header A
PABF3B	PAB EI 722 River Water Header B Break	Reactor Trip	Loss of RW header B

**PAB Elevation 735 (PABF1A, PABF1B, PABF2A, PABF2B):**

For the PAB Elevation 735' floods, an instantaneous flood rate of 6,000 to 9,000 gpm is assumed. It is assumed that the operators do not isolate the leak before a water level of 16 inches is reached. At 16 inches, the water will begin to impact the electrical equipment in MCC1-E3 and E4. Loss of these motor control centers (MCCs) would in turn prevent the operators from closing the river water header MOVs from the control room. Also, 14 inches of water would flood charging pump 1B cubicle (since it has open grating at Elevation 735') and start leaking into charging pump 1A and 1C cubicles (which have solid hatch covers). The sump pumps (10 gpm/pump) in the cubicles discharge to the North sump on Elevation 722', and would be unavailable once the water level reached 16 inches as they are powered from the MCCs. The model then gives the operators an opportunity to stop the flood before the CCR pump motors are reached at 2 feet.

Successful isolation results in the first flood scenario (floods PABF1A and PABF1B) where one river water header is unavailable, the B charging pump is unavailable, and the RWST supply to charging is unavailable.

Failure of isolation results in the second flood scenario (floods PABF2A and PABF2B) where one river water header is unavailable, the B charging pump is unavailable, all three CCR pumps are unavailable, and the RWST supply to charging is unavailable.

#### PAB Elevation 722 (PABF3A, PABF3B)

Three flood scenarios are conservatively developed for this Elevation. An instantaneous flood rate of 6,000 to 9,000 gpm is assumed for each scenario. Whereas floods on Elevation 735 required the operators to isolate the leak at the flood location, most floods (most piping and valves are upstream of isolation MOVs) on Elevation 722 can be isolated in the control room. The first flood scenario (floods PABF3A and PABF3B) results in loss of one river water header when the operators successfully isolate the leak by closing MOV-1RW-114A or MOV-1RW-114B and stopping the pump before the MOVs are flooded (leaks upstream of MOVs) at approximately 4 feet. A four-foot flood on Elevation 722 is assumed to flood the motors of MOVs at this location. This would prevent the operator from isolating a header with MOV-1RW-114A or MOV-1RW-106A on header A and MOV-1RW-114B or MOV-1RW-106B on header B. Also, the RWST supply MOVs (MOV-1CH-115B and MOV-1CH-115D) to the charging pumps are assumed flooded in the blender cubicle at the four-foot level.

#### Unit 2

The CDF for all internal floods is 1.24E-06 per year at Unit 2. A core damage frequency breakdown as a percentage of floods and total CDF for all of the Unit 2 internal floods is provided in Table 1.H-3, shown below.

**Table 1.H-3**  
Unit 2 Internal Flood Initiating Events CDF Breakdown

Rank	Flood Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Floods CDF	Percentage of Total CDF	Description
1	CVFLF	1.46E-04	6.07E-07	48.8%	2.53%	Cable Vault Flood From Fire Water
2	SGFL2	4.88E-05	3.52E-07	28.3%	1.47%	Both Safeguards Area Flood, Non-Isolated
3	SGFL1A	3.65E-04	1.11E-07	8.91%	0.46%	South Safeguards Train A Area Flood, Isolated
4	SGFL1B	3.65E-04	3.83E-08	3.08%	0.16%	North Safeguards Train B Area Flood, Isolated
5	ABFL1B	6.77E-04	2.36E-08	1.90%	0.10%	Auxiliary Building Flood, Service Water (SW) Header B, Isolated
6	VPFLB	6.77E-04	2.36E-08	1.90%	0.10%	Service Water Valve Pit Flood, Header B
7	ABFL1A	6.77E-04	2.31E-08	1.86%	0.10%	Auxiliary Building Flood, SW Header A Isolated
8	VPFLA	6.77E-04	2.31E-08	1.86%	0.10%	Service Water Valve Pit Flood, Header A
9	CVFLB	6.02E-06	1.60E-08	1.29%	0.07%	Cable Vault Flood From SW Header B
10	CVFLA	6.02E-06	1.58E-08	1.27%	0.07%	Cable Vault Flood From SW Header A
11	TBFL	7.59E-03	3.86E-09	0.31%	0.02%	Turbine Building Flood
12	CBFL	3.61E-07	2.82E-09	0.23%	0.01%	Control Building Flood
13	ISFLD	1.13E-03	1.24E-09	0.10%	0.01%	Intake Structure Flood Cube D
14	ISFLC	6.77E-04	8.33E-10	0.07%	0.00%	Intake Structure Flood Cube C
15	ISFLB	6.77E-04	4.16E-10	0.03%	0.00%	Intake Structure Flood Cube B
16	ABFL2B	2.20E-06	7.80E-11	0.01%	0.00%	Auxiliary Building Flood From SW Header B, Nonisolated
17	ABFL2A	2.20E-06	7.80E-11	0.01%	0.00%	Auxiliary Building Flood From SW Header A, Nonisolated
	<b>ALL FLOODS</b>		<b>1.24E-06</b>	<b>100%</b>	<b>5.17%</b>	

As shown in Table 1.H-3, the top six floods at Unit 2 are associated with five different flooding locations (cable vault, north & south safeguards, auxiliary building, and the service water valve pit). Together these top six floods contribute to almost 93% of the Unit 2 flood core damage frequency, but less than 5.0% of the total core damage frequency. The cable vault flood from fire water (flood CVFLF) was the top ranked

dominant flood scenario, as it contributed to almost 50% of the flood core damage frequency at Unit 2 and 2.5% of the total CDF. As such, this top flood scenario was considered to be significant enough to warrant further consideration, and was evaluated in Unit 2 SAMA-187. The cable vault flood from fire water (flood SGFL2) was the second highest ranked dominant flood scenario, as it contributed to 28% of the flood core damage frequency at Unit 2 and 1.5% of the total CDF. This flood scenario was also considered to be significant enough to warrant further consideration and was evaluated in Unit 2 SAMA-188. Each of the remaining Unit 2 floods contribute to less than 0.5% of the total CDF and were not considered to be significant enough to warrant any further consideration for SAMA evaluations.

A summary of the top six flood initiating event scenarios for Unit 2 is provided in Table 1.H-4, shown below, and the discussion that follows.

**Table 1.H-4**  
Unit 2 Top Six Internal Flood Scenarios

<b>Flood</b>	<b>Description</b>	<b>Cause of Plant Trip</b>	<b>Plant Model Impacted</b>
ABFL1B	Auxiliary Building Flood from Service Water Header B (Isolated)	Loss of Train B SW	Secondary Plant Component Cooling Water (CCS) & Train B SW
CVFLF	Cable Vault Flood from Fire Water	Loss of CCS or Reactor Trip	MCC-2-E05, E06, E13 & E14; Panels PNL-DC-10 & 11
SGFL1A	South Safeguards Train A Area Flood from Primary Plant Demineralized Water Storage Tank (PPDWST) or RWST (Isolated)	Reactor Trip	RWST, Train A Safeguards pumps, Motor-driven (MD) Auxiliary Feedwater (AFW) pump 2FWE-P23A, and Turbine-driven (TD) AFW pump 2FWE-P22
SGFL1B	North Safeguards Train B Area Flood from PPDWST or RWST (Isolated)	Reactor Trip	RWST, Train B Safeguards pumps, MD AFW pump 2FWE-P23B
SGFL2	Both Safeguards Area Flood from RWST (Non-Isolated)	Reactor Trip	RWST, Trains A & B Safeguards pumps, AFW
VPFLB	Service Water Valve Pit Flood, Header B	Loss of Train B SW	CCS & Train B SW

#### Auxiliary Building (ABFL1B):

The Auxiliary Building and Cable Vault Pipe Tunnel areas are connected at Elevation 718'-6" through a "jailhouse" door; therefore, the two areas are combined as one flood location. The Auxiliary Building at Elevation 710'-6" must fill before more than 6 inches of water can collect at the Elevation 718'-6" level. Two feet of water at Elevation 718'-6" is assumed to fail the Charging System suction and discharge valves (the charging pumps at Elevation 735'-6" are not affected).

Only three sources of water in these areas are capable of such a large volume: the RWST, Service Water System, and firewater. It is judged that service water floods dominate both the RWST and fire water risk because service water floods are found more often in the database, have large quantities of piping, cause an initiating event, and impact other support systems.

A Service Water System header failure in the Auxiliary Building will cause an initiating event due to isolation of CCS. The standby service water pump (in the header with the leak) will start, and multiple sump alarms will sound in the Auxiliary Building and/or in the Cable Vault Pipe Tunnel alerting the operators. If the leak is in one of the two headers that supply the Primary Plant Component Cooling Water System (CCP) heat exchangers, CCS, and the alternate shutdown panel, 2SWS-MOV106A and 2SWS-MOV106B can be used to isolate the headers from the control room. If the leak is in one of the two headers that supply charging pump coolers, emergency diesels, and area air conditioning units, 2SWS-MOV120A and 2SWS-MOV120B in the valve pit can isolate the headers from the control room. It is assumed that both MOVs on the header with the leak (e.g., 2SWS-MOV106B and 2SWS-MOV120B) will be isolated. The ABFL1B flood results in loss of "B" train of service water and the isolation of CCS.

#### Cable Vault and Rod Control Area (CVFLF)

A flood was postulated at Elevation 735'-6" in area CV-2 because there are several safety-related electrical cabinets, only one drain, and the flood could propagate to the adjoining area (CV-1) where redundant cabinets are located. Although doors between CV-1 and CV-2 are gasketed, they open out of the area to CV-1 and to the southeast stairwell. If the stairwell door failed first, water would spill to the pipe tunnel door that opens into the stairwell, collecting water. Another door in the stairwell at Elevation 735'-6" opens to the outside, preventing further flooding if it failed. It was conservatively assumed that a flood in area CV-2, if not isolated, would flood area CV-1 (i.e., the door to CV-1 will fail first). This flood is assumed to cause an initiating event (loss of CCS or reactor trip). The two flood sources in this area are service water and firewater. Either source is capable of flooding both cable vault areas via the propagation pathway; however, the CVFLF (firewater source flood) has a higher frequency than the service water floods.

These floods are assumed to fail the following major equipment: MCC-2-E05, MCC-2-E06, MCC-2-E13, MCC-2-E14, PNL-DC2-10, and PNL-DC2-11. Some other panels and instrumentation would also be impacted. The impact of losing this equipment due to the flood is to isolate CCS, degrade power to vital buses red and white, fail the pressurizer power-operated relief valve (PORV) block valves, prevent steam generator cooldown, prevent containment isolation, fail to makeup to the RWST from a borated water source, and fail Residual Heat Removal (RHR).

#### North and South Safeguards Areas (SGFL1A, SGFL1B, SGFL2)

The Safeguards Building houses the auxiliary feedwater pumps, quench spray pumps, low head safety injection pumps, recirculation spray pumps and heat exchangers, and piping and valves. Emergency MCCs (MCC-2-E11 and MCC-2-E12) are located at Elevations 741' and 737', respectively. The building is physically separated at Elevation 718'-6" into two separate areas, north and south, with one train of pumps in each area. Additionally, the turbine-driven auxiliary feedwater pump is located in the south safeguards area. The north area contains recirculation spray pump trains B and D, while the south area contains recirculation spray pump trains A and C. The two areas are connected by a door (no gasket) at Elevation 737'. The two areas could be connected through the RSS cubicles below Elevation 692' if the pump shaft seal fails under a flood load or if other entries are left opened or unsealed.

At Elevation 718', there is large diameter piping from the RWST and PPDWST to the safety injection, quench spray, and auxiliary feedwater pumps. The largest pipe supplied from the RWST is 14 inches in diameter, which is also the largest flood source. Any floods from these tanks into the Safeguards Areas would actuate sump alarms and the operators would be alerted to the loss of tank (RWST or PPDWST) level. Over 200,000 gallons would have to flood into the Safeguards Area to affect both the north and south areas via propagation into RSS cubicles. Since this is greater than the PPDWST volume, drain down of the PPDWST only floods one side of the safeguards area.

All pumps are approximately 2 feet off the floors. It is unlikely that any flood would be isolated before this level is reached within 2 minutes. Therefore, floods from the PPDWST or RWST (if isolated) are assumed to fail the associated pumps in their respective area, and a manual reactor trip is assumed. These events are modeled in flood scenarios SGFL1A and SGFL1B. In addition, it is conservatively assumed for these flooding scenarios that the RWST is isolated, resulting in its unavailability.

The SGFL2 flood scenario models the non-isolated RWST as being the water source, which floods both north and south safeguards areas. For this flood scenario a manual reactor trip is assumed, and all Safeguards Building pumps are failed. Additionally, the RWST is assumed to be unavailable due to its depleted volume.



### Service Water Valve Pit (VPFLB)

The Service Water Valve Pit is comprised of two separate pits that contain redundant service water motor-operated valves and instrumentation. A flood in one Service Water Valve Pit is assumed to fail one train of service water and standby service water. This scenario would lead to an initiating event, since CCS is isolated on low pressure in either header. Failure of one train of service water ultimately results in the loss of one train of high head safety injection, CCP, and an emergency diesel generator. A severe flood would fill the pit and pipe tunnel, eventually discharging into the yard as concrete plugs would be pushed up by the water. The operators would isolate the flood by stopping service water pumps and by closing motor-operated valves in the Intake Structure. Flood scenario VPFLB models flooding in the B valve pit.

## Question SAMA-2

Provide the following information relative to the Level 2 PRA analysis:

- a. According to Table 3.4.3-2, there are ten release categories plus an intact containment category which are used for the SAMA case runs. Provide a description of the fission product release fractions used for the MACCS2.
- b. It is noted that MAAP-DBA was used to support the Level 2 analysis rather than the more widely-used MAAP 4.0.4 computer code. Discuss the rationale for using MAAP-DBA and the estimated impact of this code choice on accident progression and source terms. Provide a comparison of the source terms obtained using MAAP-DBA with source terms based on MAAP 4.0.4 for comparable sequences.
- c. In Section 3.2.1 it is stated that it is not necessary to run a MAAP-DBA case to represent each individual release class. For example, for Release Type I, release categories BV1, BV3, BV18, and BV19 were re-analyzed, but BV2 and BV4 were not. Explain why the MAAP-DBA reanalysis was performed for only a subset of the release categories, how this subset was selected, and how the remaining release categories were treated.

### RESPONSE SAMA-2.a

- a. According to Table 3.4.3-2, there are ten release categories plus an intact containment category which are used for the SAMA case runs. Provide a description of the fission product release fractions used for the MACCS2.

The fission product release fractions used in MACCS2 for each release category are shown below in Table 2.A-1, along with the representative Level 2 release category accident sequence bins. These release fractions were developed based on the fourteen BVPS Level 2 release categories reanalyzed using MAAP-DBA. A description of the representative MAAP-DBA accident sequences by release category (bin) are provided in Unit 1 and Unit 2 Tables 3.2.1-6 of the ER, Attachments C-1 and C-2, pages C.1-34 and C.2-34.

These MAAP-DBA accident sequences were analyzed for both BVPS Units, and a composite set of data was generated by taking the maximum release fraction between the two Units. When multiple Level 2 release categories were used to define the ten MACCS2 release categories plus an intact containment category for the SAMA case runs, the bounding Level 2 release category was used. The fission product release fractions shown in Table 2.A-1 were used in analyzing both Unit 1 and Unit 2 SAMAs.

**Table 2.A-1**  
Fission Product Release Fractions Used in MACCS2

MACCS2 Release Category	Level 2 Release Category	Plume No.	FISSION PRODUCT RELEASE FRACTIONS								
			NG	I	Cs	Te	Ba	Sr	La	Mo	Ce
INTACT	BV21	1	9.73E-05	3.90E-07	3.60E-07	8.00E-08	5.23E-08	3.60E-08	3.28E-08	1.01E-07	4.69E-10
	BV21	2	9.43E-04	5.30E-08	9.45E-09	7.44E-11	0.00E+00	0.00E+00	1.00E-11	0.00E+00	2.86E-13
<b>ECF</b>											
VSEQ	BV19	1	1.00E+00	1.73E-01	1.71E-01	1.48E-02	2.42E-02	1.32E-02	2.00E-04	4.88E-02	1.77E-05
SGTR	BV18	1	1.00E+00	4.22E-01	2.81E-01	1.57E-02	1.15E-02	1.73E-03	7.39E-05	3.40E-02	3.73E-06
DCH	BV1 BV3	1	1.00E+00	2.07E-01	1.62E-01	1.07E-01	1.66E-02	2.19E-02	1.32E-02	1.46E-02	5.70E-04
<b>SECF</b>											
VSEQ	BV20	1	1.00E+00	7.65E-02	7.23E-02	6.12E-03	8.68E-03	6.20E-03	1.95E-04	1.31E-02	1.93E-05
LOCI	BV7	1	2.74E-02	5.90E-03	5.89E-03	2.74E-04	2.13E-04	2.06E-04	3.17E-06	6.46E-05	4.30E-07
	BV7	2	9.73E-01	2.66E-02	2.44E-02	1.40E-02	1.83E-03	1.62E-03	4.58E-05	2.90E-03	2.10E-05
BV5	BV5	1	2.93E-03	2.73E-04	1.83E-04	2.53E-06	8.62E-07	1.40E-07	8.45E-09	4.47E-07	5.13E-10
	BV5	2	9.77E-01	7.67E-03	3.83E-03	9.95E-03	3.42E-03	3.41E-03	3.41E-03	3.48E-03	4.96E-05
<b>LATE</b>											
Large	BV10 BV12	1	4.09E-01	1.39E-03	1.90E-04	8.55E-06	1.82E-07	1.81E-07	1.81E-07	2.12E-07	2.59E-09
	BV10 BV12	2	8.07E-02	4.30E-05	1.99E-05	3.61E-07	1.22E-09	7.40E-10	1.50E-11	4.50E-09	2.70E-12
Small	BV13 BV15	1	1.43E-01	9.67E-04	3.28E-04	1.13E-04	2.05E-06	1.69E-06	7.74E-07	5.98E-06	2.93E-08
	BV13 BV15	2	8.64E-01	3.29E-02	7.93E-03	1.81E-03	3.86E-06	2.44E-06	1.02E-07	2.70E-06	7.71E-08
H2 Burn	BV9	1	5.02E-01	4.31E-03	8.60E-04	9.40E-03	4.14E-05	4.68E-05	2.54E-05	2.30E-05	1.69E-06
	BV9	2	4.98E-01	1.42E-02	3.99E-03	1.79E-02	2.68E-05	2.19E-05	1.17E-05	1.11E-05	1.03E-06
BMMT	BV17	1	1.00E+00	1.41E-02	4.19E-03	1.34E-03	4.65E-06	8.04E-07	7.87E-07	8.44E-07	1.93E-08

RESPONSE SAMA-2.b

- b. It is noted that MAAP-DBA was used to support the Level 2 analysis rather than the more widely-used MAAP 4.0.4 computer code. Discuss the rationale for using MAAP-DBA and the estimated impact of this code choice on accident progression and source terms. Provide a comparison of the source terms obtained using MAAP-DBA with source terms based on MAAP 4.0.4 for comparable sequences.**

The MAAP-DBA computer code is the BVPS current licensing basis analysis tool for determining the containment response to a design basis accident (DBA); it is based on MAAP 4.0.5. A detailed discussion of the MAAP-DBA code is docketed in FENOC Letter L-03-188 (ADAMS accession number ML033350145) to the NRC, dated November 24, 2003. The MAAP-DBA code was developed for the BVPS atmospheric containment conversion (ACC) program, and approval was granted by the NRC in Amendments 271 and 153 for Unit 1 and Unit 2. In support of this ACC program, single and multiple node containment models were developed for both units. The single node containment models are used for determining the DBA peak containment pressure, gas temperature, and liner temperature; while the multiple node containment models are used for other DBA containment response attributes (e.g., sump temperature) and MAAP-DBA analyses, including the BVPS SAMA analyses. The containment models developed for MAAP-DBA are not backwards compatible; hence, a direct comparative analysis with MAAP 4.0.4 is not possible without significant modifications to the current plant models. However, for comparative purposes it was practicable to modify the Unit 2 multiple node containment model to be compatible with MAAP 4.0.4 in order to provide confirmation that MAAP-DBA yields results that are either bounding or in reasonable agreement with the more widely-used MAAP 4.0.4 version. The discussion below summarizes the approach taken to provide a meaningful comparison of source terms for the two versions of the code.

The MAAP-DBA computer code, as well as all versions of MAAP 4, can be separated into several main regions. These regions are listed below:

- Containment
- Engineered Safeguards (containment & primary system)
- Primary System
- Core & Fission Products

The engineered safeguards, primary system, and core & fission products models within the MAAP-DBA computer code are essentially the same as those contained in MAAP 4.0.5; therefore, the differences in source terms from the degraded/failed fuel would only be driven by differences between the MAAP 4.0.4 and MAAP 4.0.5 codes. These differences are not considered significant, but could result in slight variations in the accident progression and timing of core heat-up/melt, melt progression, fission

product deposition in the primary system, vessel failure, and core debris distribution (i.e., in the vessel, cavity, or dispersed into the lower compartment via a high pressure melt ejection) following vessel failure. All of these factors can contribute to differences in fission product release fractions through changes in the fission product inventories released into the containment, the deposition processes in the containment and the subsequent releases to the environment following containment failure.

The primary difference in the containment models between the MAAP-DBA and MAAP 4.0.4 versions of the code is that the MAAP-DBA code has enhanced containment modeling. Table 2.B-1, shown below, provides a brief summary of the enhancements.

**Table 2.B-1**  
Summary of MAAP 4.0.4 and MAAP-DBA Differences

	<b>MAAP 4.0.4</b>	<b>MAAP-DBA</b>
Paint on Containment Heat Sinks	Not modeled	Up to 4 separate layers of paint can be modeled
Nodalization of Containment Heat Sinks	Coarse nodalization (no more than 20 nodes)	Finer nodalization (up to 40 nodes)
Modeling of Containment Liner	Modeled as a single node	Modeled as several nodes
Tagami & Uchida heat transfer correlations	Not modeled	Available for single node containment calculations
Containment Metal Heat Sinks	Modeled as a single node	Modeled as several nodes
Natural Convection Heat Transfer Coefficient (as a function of the Grashof number)	Grashof number = $f(\Delta T/T)$	Grashof number = $f(\Delta \rho/\rho)$

From a source term perspective, the primary effects of the enhancements will cause heat sink temperatures to be different, thus affecting the thermophoresis and diffusiphoresis processes. Additionally, these enhancements can cause differences in the overall containment response in terms of pressure and temperature which would affect fission product deposition rates and release rates.

To provide a reasonable comparison, four separate release classes using the modified BVPS Unit 2 multiple node containment model were run using the MAAP 4.0.4 computer code. The Unit 2 Level 2 release categories reexamined were BV1, BV3, BV5 and BV9 as identified in Table 3.2.1-6 of the ER, Attachment C-2, page C.2-34. The decision for choosing these comparison sequences was based on release categories that would illustrate meaningful comparisons of containment performance of

the MAAP-DBA and MAAP 4.0.4 containment models, and not necessarily the highest release frequencies. The release categories chosen for evaluation provide comparison results for both early and late containment failure times and releases for both high and low pressure vessel failure scenarios.

The results of these comparisons in terms of fission product release fractions are illustrated in Figures 2.B-1a through 2.B-4b, shown below, for the major fission product groups (i.e., noble gas constituents, cesium iodine (CsI), cesium hydroxide (CsOH), and a representative non-volatile fission product - strontium oxide (SrO)). As shown in these figures, the comparison between the two computer code results for the four cases either demonstrate good agreement or illustrate that the results generated using the MAAP-DBA computer code are bounding. For the most part, the release fractions followed the same trend and the release fractions were on the same order of magnitude and typically differ by no more than a factor of 2 to 3. It is important to note that these differences occur when the release fractions are relatively small such that the differences can be due to small differences in the models.

The notable difference in the slope of the CsOH release results for two BV9 cases (shown on Figure 2.B-4b) are due to code differences in the primary system and core sections, as well as the differences in the containment models (each of which could have some influence on the observed differences). A review of the output for the BV9 CsOH release revealed that the vessel failure timings are different by about 0.5 hours (MAAP 4.0.4 predicted a later vessel failure), and the fission product distribution following vessel failure is different (with MAAP 4.0.4 retaining more fission products in the primary system and the corium and less available in the containment gas space to be released upon containment failure).

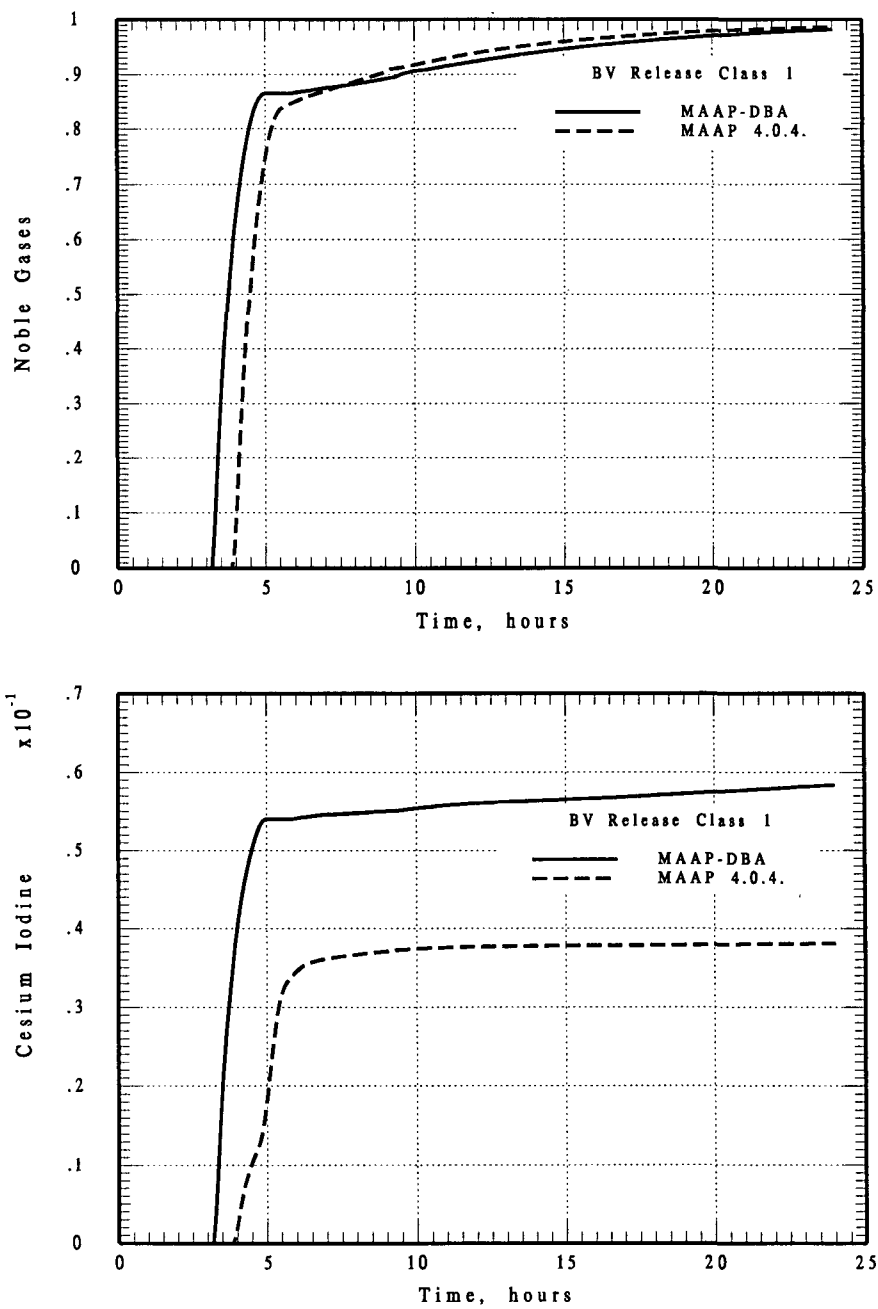
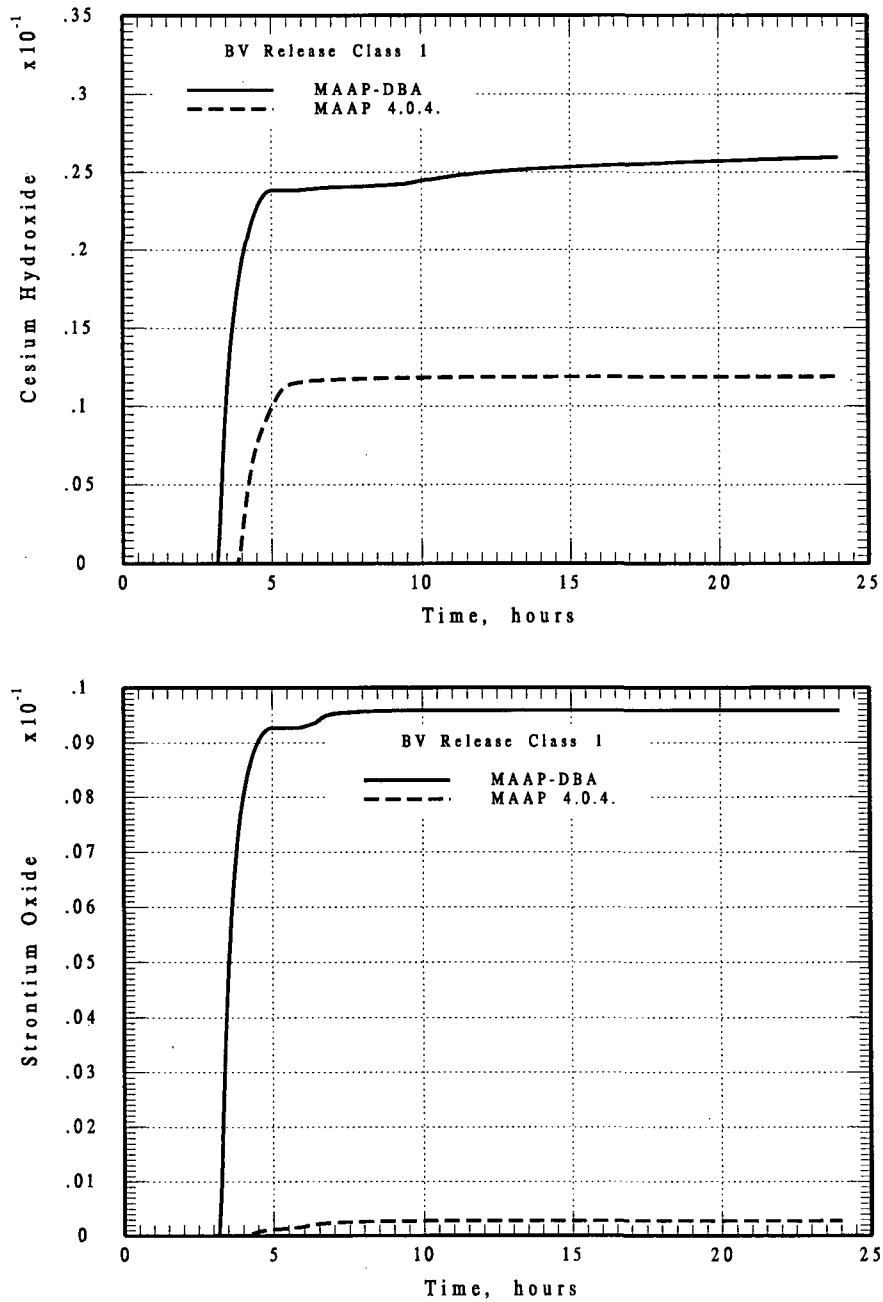


Figure 2.B-1a: Noble Gas and Cesium Iodine release fractions for BVPS Unit 2 Release Class BV1



**Figure 2.B-1b:** Cesium Hydroxide and Strontium Oxide release fractions for BVPS Unit 2 Release Class BV1



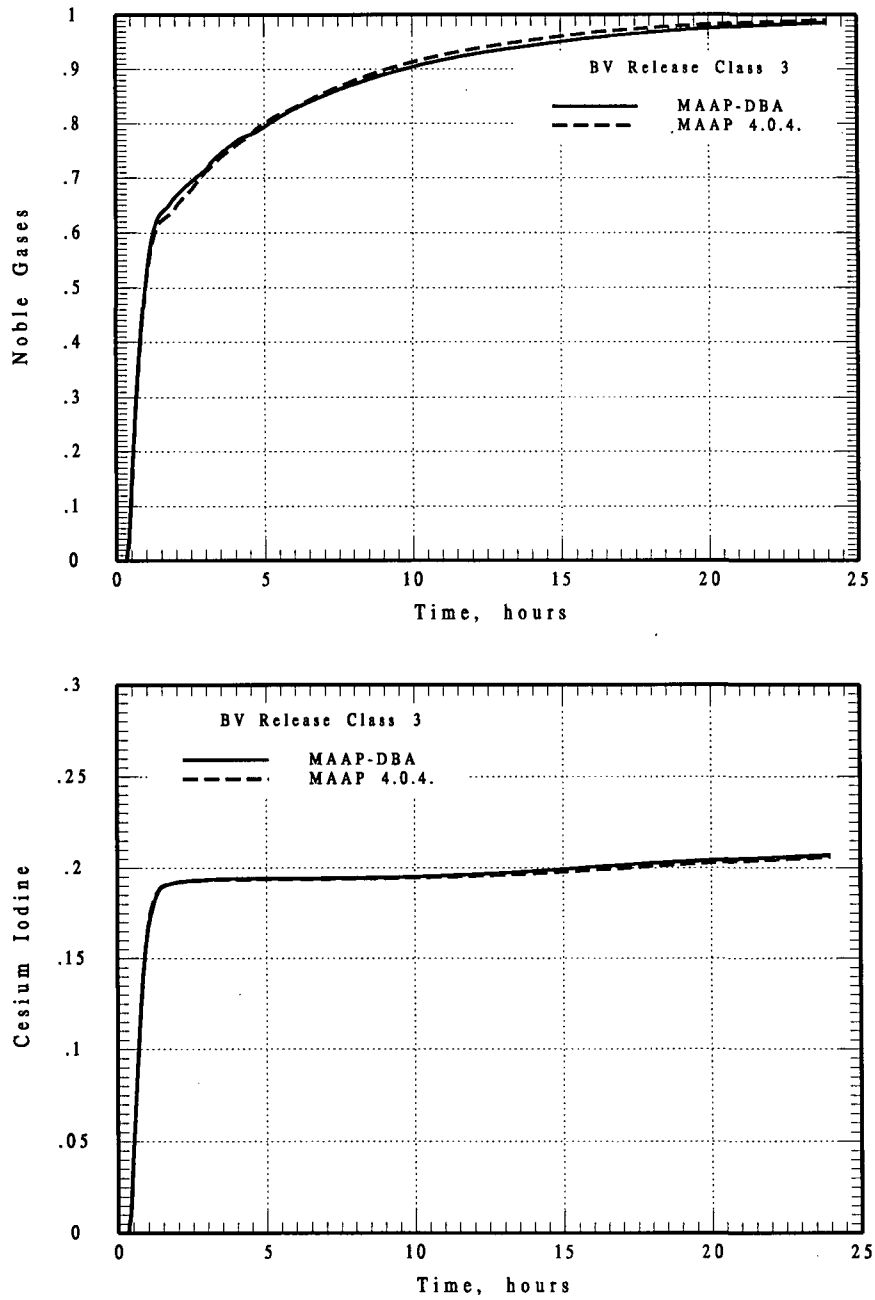


Figure 2.B-2a: Noble Gas and Cesium Iodine release fractions for BVPS Unit 2 Release Class BV3

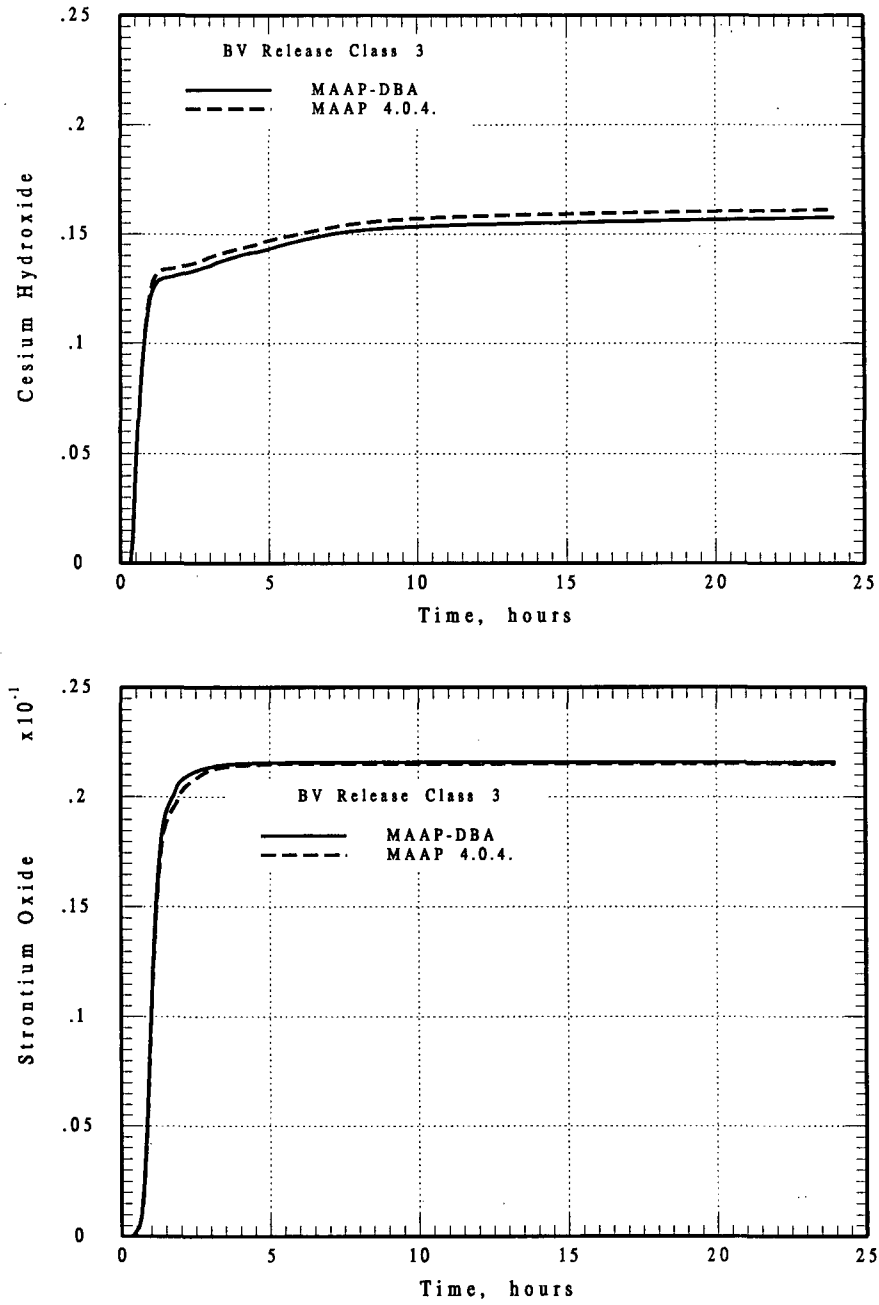


Figure 2.B-2b: Cesium Hydroxide and Strontium Oxide release fractions for BVPS Unit 2 Release Class BV3

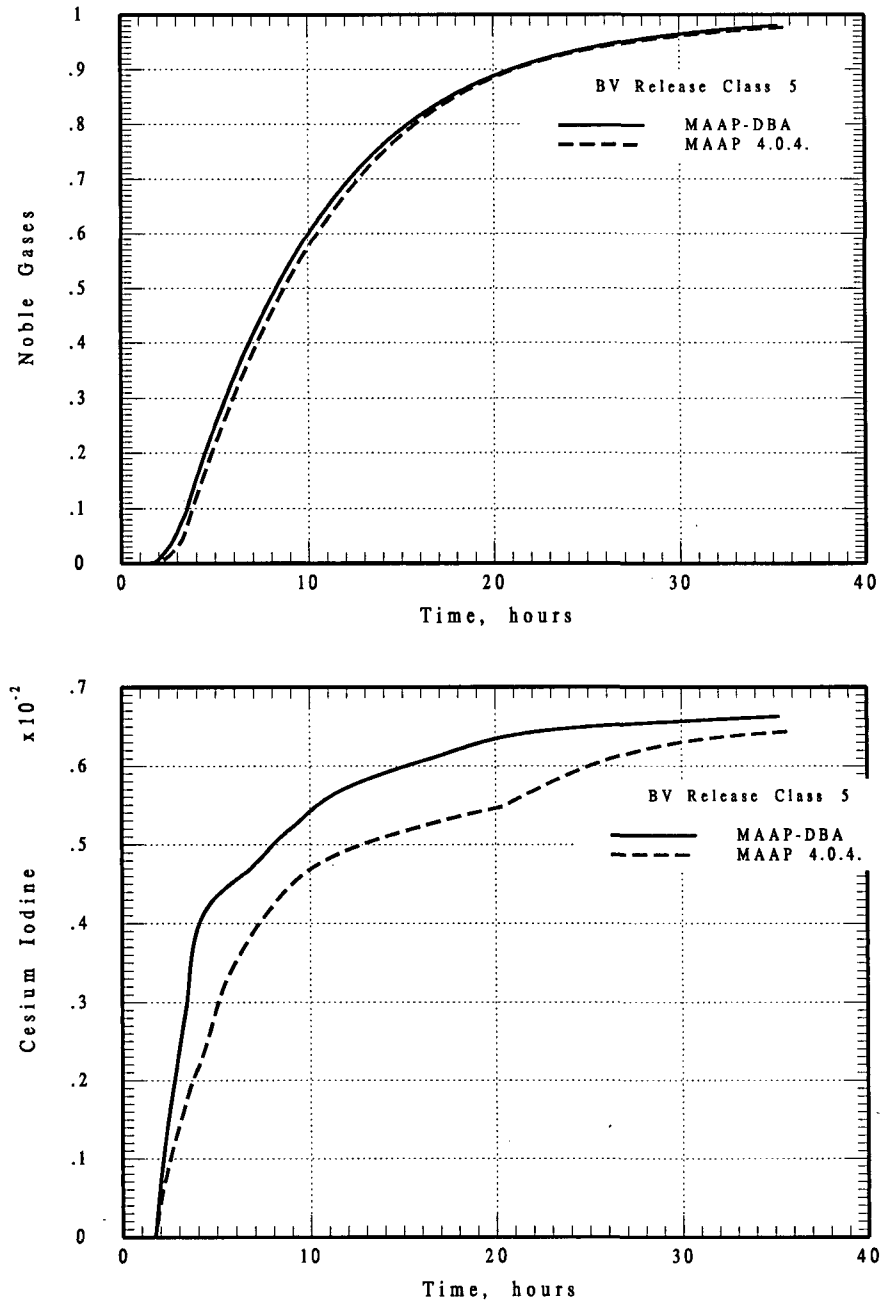


Figure 2.B-3a: Noble Gas and Cesium Iodine release fractions for BVPS Unit 2 Release Class BV5

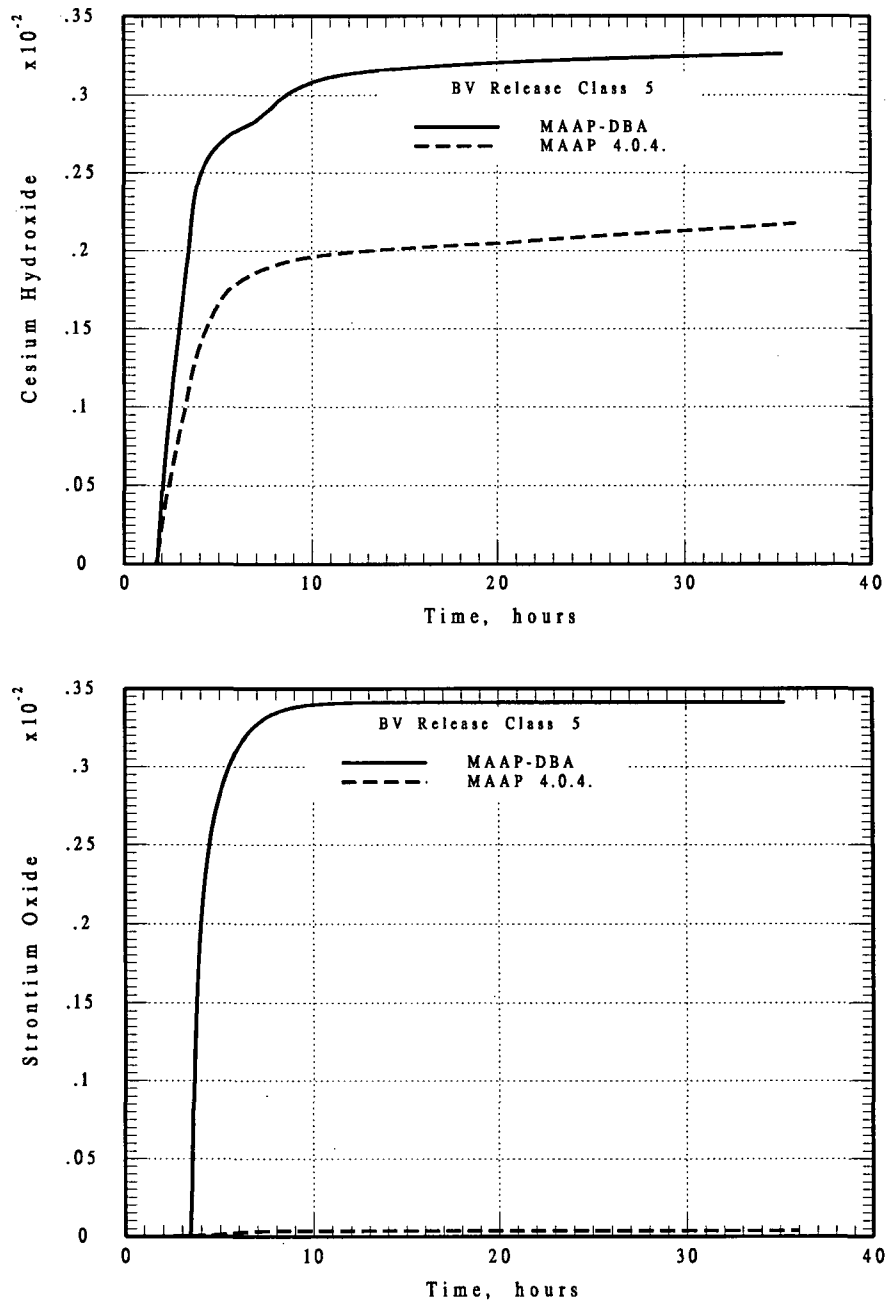
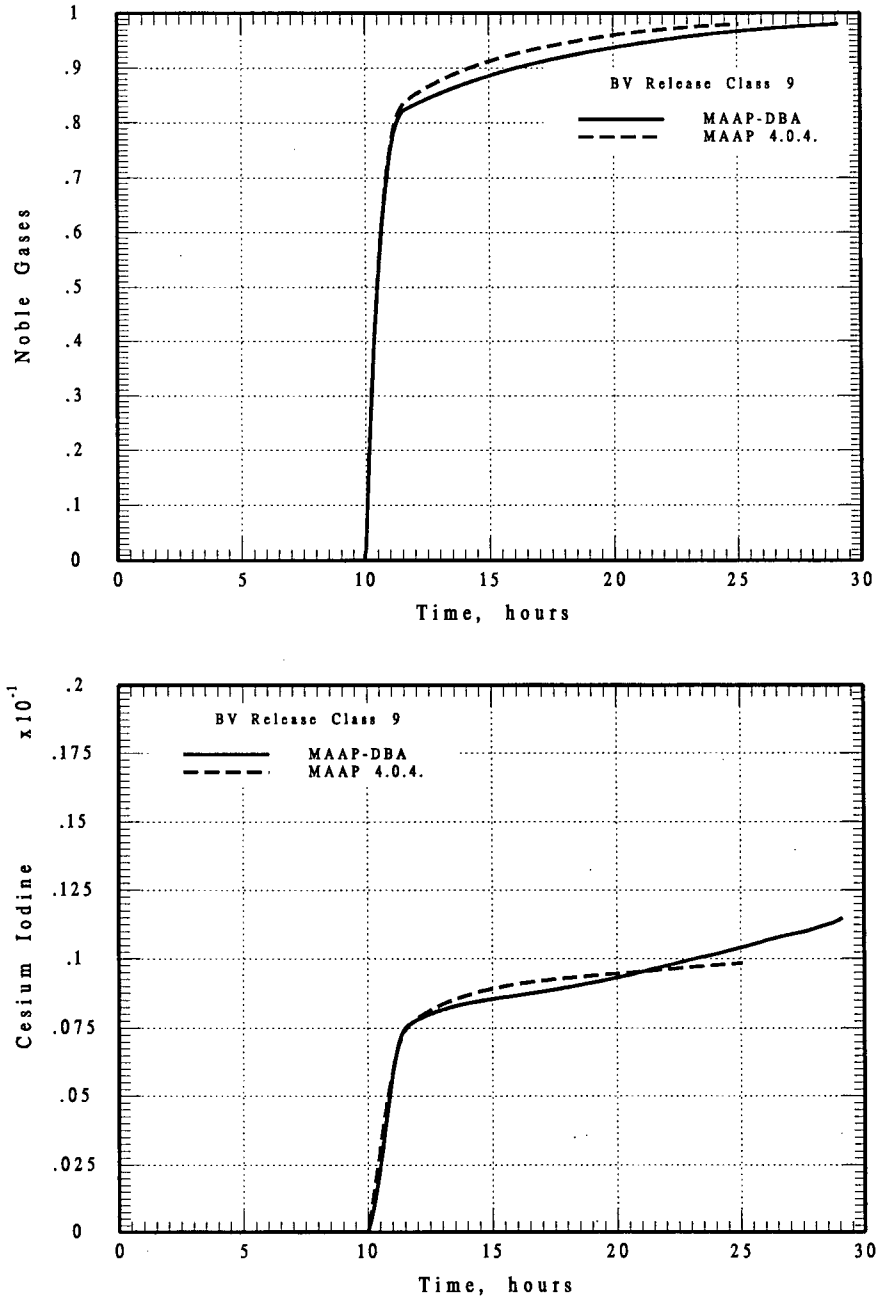
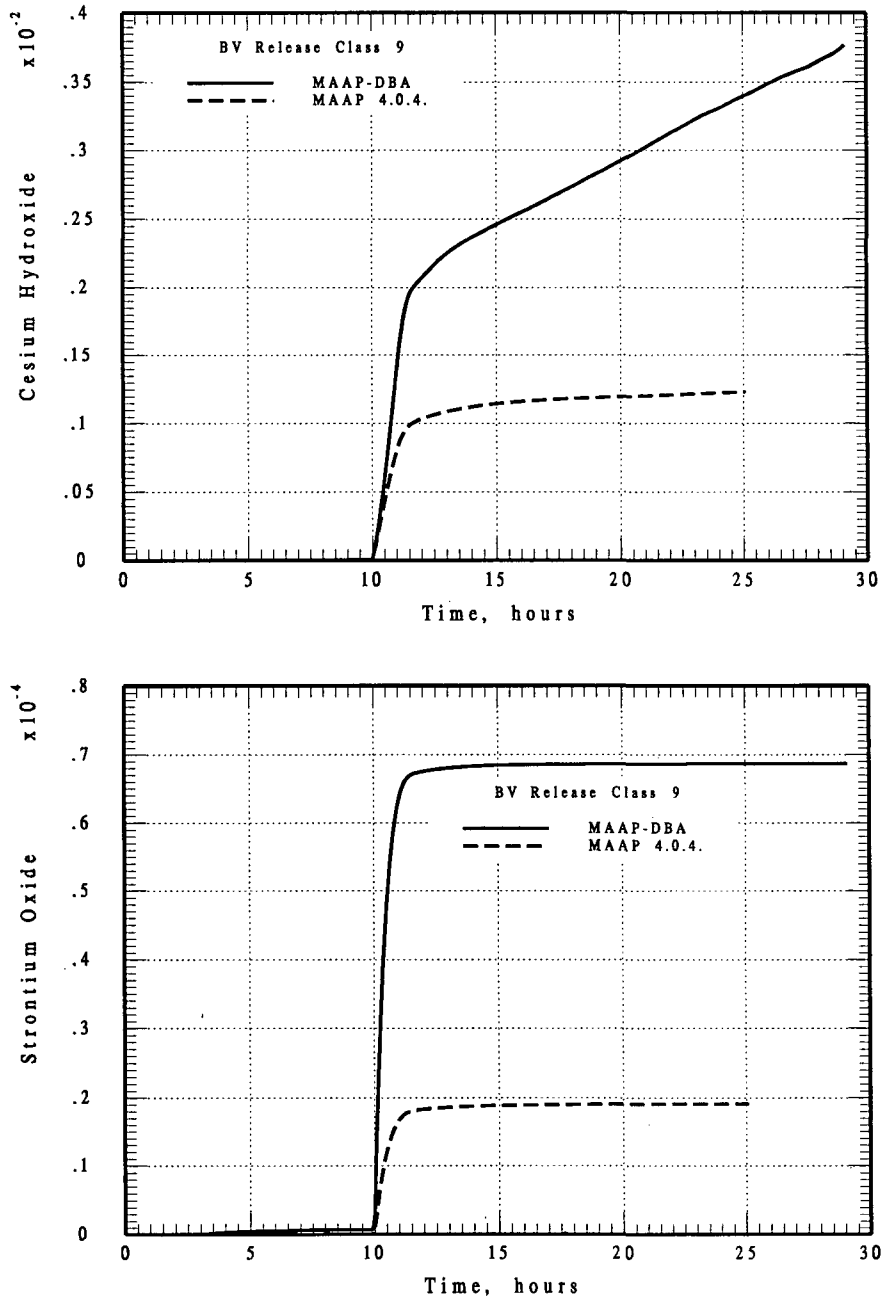


Figure 2.B-3b: Cesium Hydroxide and Strontium Oxide release fractions for BVPS Unit 2 Release Class BV5.



**Figure 2.B-4a:** Noble Gas and Cesium Iodine release fractions for BVPS Unit 2 Release Class BV9



**Figure 2.B-4b:** Cesium Hydroxide and Strontium Oxide release fractions for BVPS Unit 2 Release Class BV9

RESPONSE SAMA-2.c

- c. In Section 3.2.1 it is stated that it is not necessary to run a MAAP-DBA case to represent each individual release class. For example, for Release Type I, release categories BV1, BV3, BV18, and BV19 were re-analyzed, but BV2 and BV4 were not. Explain why the MAAP-DBA reanalysis was performed for only a subset of the release categories, how this subset was selected, and how the remaining release categories were treated.**

The twenty-one BVPS PRA Level 2 release categories (BV1 through BV21) are the categories defined for the original IPE analysis, and are presented in Unit 1 and Unit 2 Tables 3.2.1-5 of the ER, Attachments C-1 and C-2, pages C.1-33 and C.2-33. In developing the fission product release fractions used for the SAMA evaluations, it was judged that not all of these release categories had to be reanalyzed using MAAP-DBA, due to the fact that their source term release fractions are effectively bounded by the fourteen representative cases that were reanalyzed. These fourteen representative cases are described in Unit 1 and Unit 2 Tables 3.2.1-6 of the ER, Attachments C-1 and C-2, pages C.1-34 and C.2-34.

The first sixteen release categories provided on Table 3.2.1-5 (i.e., BV1 through BV16) are grouped in pairs with similar RCS pressure and containment failure characteristics. For each pair, the only difference in characteristics between the two release categories is the operating status of the containment heat removal spray system. For example, release category BV1 is a large, early containment failure with a high RCS pressure and no containment sprays operable, while release category BV2 is a large, early containment failure with a high RCS pressure and containment sprays operable. Since these paired release categories are similar, the IPE fission product release fractions were used as selection criteria to limit the number of Level 2 release categories that were reanalyzed using MAAP-DBA. In general, whichever of the paired release categories produced larger fission product release fractions in the IPE was the one selected to be reanalyzed for the SAMA evaluation. An exception to this was for the Major Release Type III large, late containment failures. For this subset, release category BV12 was selected, since its paired release category BV11 bin had zero frequency for the baseline and all of the SAMA sensitivity cases that were evaluated. In addition, the pair of release categories BV9 and BV10 were both reanalyzed, due to being binned into different SAMA release categories as shown in Unit 1 and Unit 2 Tables 3.4.3-2 of the ER, Attachments C-1 and C-2, pages C.1-50 and C.2-51. Furthermore, the last five release categories (BV17 through BV21) are significantly different enough in both release characteristics and fission product release fractions that each was reanalyzed with MAAP-DBA.

This process resulted in eliminating the MAAP-DBA reanalysis for release categories BV2, BV4, BV6, BV8, BV11, BV14, and BV16, since they are effectively bounded by either the fission product release fractions of the fourteen representative release categories that were reanalyzed, or had zero release frequency. The fourteen release

categories reanalyzed with MAAP-DBA were then combined as discussed in the response to RAI 2.a to obtain the ten release categories plus intact containment category used in the SAMA evaluations, and presented in Table 3.4.3-2.



### Question SAMA-3

Provide the following information regarding the treatment of external events in the SAMA analysis:

- a. **Confirm which versions of the internal events PRA were used to develop the fire CDF values reported in Table 3.1.2.1-1. Provide a summary of the dominant fire scenarios for the current fire model in terms of overall fire frequency, plant initiator, and structures, systems, and components (SSCs) impacted.**
- b. **Provide a summary of the dominant seismic scenarios for the current seismic model in terms of overall seismic initiator frequency, plant initiator, and SSCs impacted.**

#### RESPONSE SAMA-3.a

- a. **Confirm which versions of the internal events PRA were used to develop the fire CDF values reported in Table 3.1.2.1-1. Provide a summary of the dominant fire scenarios for the current fire model in terms of overall fire frequency, plant initiator, and structures, systems, and components (SSCs) impacted.**

The current Revision 4 PRA models (BV1REV4 and BV2REV4) are fully integrated PRA models that include both internal and external events, as well as both the Level 1 and the Level 2 risk models. These PRA models were used to develop the fire CDF values of 3.67E-06 per year at Unit 1 and 4.80E-06 per year at Unit 2, as reported in Tables 3.1.2.1-1 of the SAMA report. A summary of the top ten ranking dominant fire scenarios and their core damage frequency breakdown as a percentage of fires and total CDF for each Unit is provided in Tables 3.A-1 (Unit 1) and 3.A-3 (Unit 2). Tables 3.A-2 (Unit 1) and 3.A-4 (Unit 2) provide a summary for each of the Unit's top ten fire scenarios in terms of overall fire frequency, plant initiator (i.e., cause for the plant tripping following the fire), and the structures, systems, and components (SSCs) impacted by the fire that are modeled in the PRA.

Tables 3.A-1 (Unit 1), 3.A-2 (Unit 1), 3.A-3 (Unit 2), and 3.A-4 (Unit 2) are shown below.

**Table 3.A-1**  
Unit 1 Top 10 Dominant Fire Initiating Events CDF Summary

Rank	Fire Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Fire CDF	Percentage of Total CDF	Fire Scenario Description
1	CV3L1A	3.10E-06	6.30E-07	17.2%	3.2%	Cable tunnel fire initiated by cable in trays at northernmost end of room that damages cables from east wall to west wall at the north end of room
2	CR1L1P	1.47E-05	4.59E-07	12.5%	2.4%	Control room cable fire in Benchboard C, Sections C1, C2, & C3
3	CS1L1E	1.77E-06	3.61E-07	9.8%	1.9%	Cable spreading area cable fire in northeast corner trays that damages two stacks of cable trays running side-by-side
4	CS1L1C	3.27E-05	2.81E-07	7.7%	1.4%	Cable spreading area with one of the 3 clustered emergency switchgear HVAC fans igniting that destroys one of the other nearby fans
5	CV3L1B	1.15E-06	2.34E-07	6.4%	1.2%	Cable tunnel fire initiated by cable in trays just south of northernmost end of room that damages cables from east wall to west wall at the north end of room
6	CR4L1C	1.92E-07	1.92E-07	5.2%	1.0%	Process rack room normal battery (BAT-5) fire that damages vertical cable trays at middle of south wall
7	CR1L1O	5.71E-06	1.79E-07	4.9%	0.9%	Control room cable fire in Benchboard C, Sections C1, C2
8	CR3L1E	7.73E-07	1.57E-07	4.3%	0.8%	Relay panel room fire initiated by one of three logic cabinets in middle of room, damaging cables running across the south end of room
9	CR4L1D	1.14E-07	1.14E-07	3.1%	0.6%	Process rack room normal battery charger fire with > 30' radius that damages vertical cable trays at middle of south wall
10	CR4L1E	5.57E-07	1.13E-07	3.1%	0.6%	Process rack room normal battery charger fire with > 8' radius that damages power cables for auxiliary river water pumps and control cables for main river water pump valves
<b>Totals for Top 10 Dominant Fires</b>			<b>2.72E-06</b>		<b>13.9%</b>	
<b>Totals for All Fires</b>			<b>3.67E-06</b>		<b>18.8%</b>	

**Table 3.A-2**  
Unit 1 Top 10 Dominant Fire Scenarios Impact Summary

<b>Fire Initiating Event</b>	<b>Overall Fire Frequency</b>	<b>Plant Initiator</b>	<b>PRA Modeled SSCs Impacted</b>
CV3L1A	3.10E-06	Reactor trip from loss of river water	River water trains A & B, and turbine plant component cooling water
CR1L1P	1.47E-05	Reactor Trip from loss of main feedwater and instrument air	Main feedwater, auxiliary feedwater, turbine plant component cooling water, main steam, station instrument air, and containment instrument air
CS1L1E	1.77E-06	Reactor trip from loss of river water	River water trains A & B, and containment isolation
CS1L1C	3.27E-05	Manual reactor trip from loss of emergency switchgear ventilation	Normal and emergency switchgear ventilation, with credit for operators to start portable fans and open doors
CV3L1B	1.15E-06	Reactor trip from loss of river water	River water trains A & B, and turbine plant component cooling water
CR4L1C	1.92E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., main feedwater, auxiliary feedwater, emergency AC power, Engineered Safety Features (ESF) equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CR1L1O	5.71E-06	Reactor Trip from loss of main feedwater and instrument air	Main feedwater, auxiliary feedwater, turbine plant component cooling water, and station instrument air
CR3L1E	7.73E-07	Reactor trip from loss of river water	River water trains A & B
CR4L1D	1.14E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., main feedwater, auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CR4L1E	5.57E-07	Reactor trip from loss of river water	River water trains A & B

**Table 3.A-3**

**Unit 2 Top 10 Dominant Fire Initiating Events CDF Summary**

Rank	Fire Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Fire CDF	Percentage of Total CDF	Fire Scenario Description
1	DG1L1A	1.04E-02	9.12E-07	19.0%	3.8%	#1 emergency diesel generator building fire from any source in area that damages the #1 EDG
2	DG2L1A	1.03E-02	9.06E-07	18.9%	3.8%	#2 emergency diesel generator building fire from any source in area that damages the #2 EDG
3	CT1L1A	5.08E-07	5.08E-07	10.6%	2.1%	Cable tunnel human error (transient combustibles) fires that damage both orange and purple cables in southeast corner of room
4	CT1L1B	3.05E-07	3.05E-07	6.4%	1.3%	Cable tunnel human error (transient combustibles) fires that damage both orange & purple cables along north wall of room
5	CB3L1P	1.76E-05	2.55E-07	5.3%	1.1%	Control room cable fire in Benchboard C, Sections C1, C2, & C3
6	SB0P4A	2.49E-07	2.49E-07	5.2%	1.0%	Battery room 2-5 fire from any source in area with > 20' radius that propagates to the west cable vault, primary auxiliary building, and normal switchgear area that damages both orange & purple cables in all four fire zones
7	CB1P1A	2.47E-07	2.47E-07	5.2%	1.0%	Relay panel room fire from any source in area with > 20' radius that propagates to the cable tunnel damages both orange & purple cables in both areas
8	CV3L1F	1.82E-04	1.78E-07	3.7%	0.7%	West cable vault elevation 755' fire in logic cabinet 2MSS-HYV101A, B, & C that damages multiple purple train cable trays
9	CT1P2A	1.53E-07	1.53E-07	3.2%	0.6%	Cable tunnel fire from any source in area with > 20' radius that propagates to the cable spreading room and damages both orange & purple cables in both areas
10	CT1P1A	1.53E-07	1.53E-07	3.2%	0.6%	Cable tunnel fire from any source in area with > 20' radius that propagates to the relay panel room and damages both orange & purple cables in both areas
<b>Totals for Top 10 Dominant Fires</b>			<b>3.87E-06</b>		<b>16.1%</b>	
<b>Totals for All Fires</b>			<b>4.80E-06</b>		<b>20.0%</b>	

**Table 3.A-4**

Unit 2 Top 10 Dominant Fires Scenarios Impact Summary

Fire Initiating Event	Overall Fire Frequency	Plant Initiator	SSCs Impacted
DG1L1A	1.04E-02	Manual Reactor Trip due to loss of EDG	#1 emergency diesel generator
DG2L1A	1.03E-02	Manual Reactor Trip due to loss of EDG	#2 emergency diesel generator
CT1L1A	5.08E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CT1L1B	3.05E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., main feedwater, auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CB3L1P	1.76E-05	Reactor Trip from loss of main feedwater and instrument air	Main feedwater, auxiliary feedwater, secondary plant component cooling water, station instrument air, containment instrument air, and main steam
SB0P4A	2.49E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., main feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CB1P1A	2.47E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., main feedwater, auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CV3L1F	1.82E-04	Reactor Trip due to loss of purple train of emergency AC power	Emergency AC purple power, reactor plant component cooling water, secondary plant component cooling water, and containment instrument air
CT1P2A	1.53E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.
CT1P1A	1.53E-07	Reactor Trip	Virtually all PRA modeled systems are impacted (e.g., auxiliary feedwater, emergency AC power, ESF equipment, and river water). Therefore, this scenario is modeled as going directly to core damage.

In identifying potential SAMAs for fires, only those fire scenarios that have a greater than 1% contribution to the total CDF were considered. It was judged that the risk associated with fire scenarios having a 1% or less contribution to the total CDF do not significantly impact the core damage frequency, and as such do not warrant any SAMA evaluation considerations. Therefore, the top five ranking dominant fire scenarios presented in Table 3.A-1 for Unit 1 and Table 3.A-3 for Unit 2 were considered for potential SAMAs.

At Unit 1, the top five ranking dominant fire scenarios include the following fire scenarios and SAMA Numbers:

- CV3L1A – SAMA 180,
- CR1L1P – SAMA 190 (See Enclosure B for ER Unit 1 SAMA 190 information),
- CS1L1E – SAMA 183,
- CS1L1C – SAMAs 143, 168 & 181, and
- CV3L1B – SAMA 180.

Of these, only SAMA 168 was determined to be cost beneficial in reducing the risk associated with fires originating from the HVAC fans located in the cable spreading area; all other fire scenario evaluations had costs exceeding the benefits.

At Unit 2, the top five ranking dominant fire scenarios include the following fire scenarios and SAMA Numbers:

- DG1L1A – SAMA 184,
- DG2L1A – SAMA 185,
- CT1L1A – SAMA 180,
- CT1L1B – SAMA 180, and
- CB3L1P – SAMA 179.

All of these fire scenario evaluations had costs exceeding the benefits.

#### RESPONSE SAMA-3.b

- b. Provide a summary of the dominant seismic scenarios for the current seismic model in terms of overall seismic initiator frequency, plant initiator, and SSCs impacted.**

The CDF for all seismic events is  $1.19\text{E-}05$  per year at Unit 1 and  $9.70\text{E-}06$  per year at Unit 2, as reported in Unit 1 and Unit 2 Tables 3.1.2.2-1 of the ER, Attachments C-1 and C-2, pages C.1-23 and C.2-22. A summary of the seismic initiating event frequencies

and their core damage frequency breakdown as a percentage of seismic events and total CDF for each Unit is provided in Tables 3.B-1 (Unit 1) and 3.B-2 (Unit 2), shown below.

**Table 3.B-1**  
Unit 1 Seismic Initiating Events CDF Summary

Rank	Seismic Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Seismic CDF	Percentage of Total CDF	Seismic Scenario Description
1	SEIS1	1.42E-04	5.04E-07	4.2%	2.6%	Earthquakes with a 0.01g - 0.25g peak ground acceleration
2	SEIS2	1.70E-05	3.09E-06	26.0%	15.8%	Earthquakes with a 0.25g - 0.35g peak ground acceleration
3	SEIS3	8.36E-06	5.29E-06	44.5%	27.1%	Earthquakes with a 0.35g - 0.50g peak ground acceleration
4	SEIS4	2.93E-06	2.93E-06	24.7%	15.0%	Earthquakes with a 0.50g - 1.00g peak ground acceleration
5	SEIS5	7.58E-08	7.58E-08	0.6%	0.4%	Earthquakes with a 1.00g - 1.33g peak ground acceleration
<b>Totals for All Seismic Events</b>			<b>1.19E-05</b>		<b>60.9%</b>	

**Table 3.B-2**  
Unit 2 Seismic Initiating Events CDF Summary

Rank	Seismic Initiating Event	Initiating Event Frequency	Core Damage Frequency	Percentage of Seismic CDF	Percentage of Total CDF	Seismic Scenario Description
1	SEIS1	1.42E-04	2.20E-07	2.3%	0.9%	Earthquakes with a 0.01g - 0.25g peak ground acceleration
2	SEIS2	1.70E-05	2.02E-06	20.8%	8.4%	Earthquakes with a 0.25g - 0.35g peak ground acceleration
3	SEIS3	8.36E-06	4.46E-06	46.0%	18.6%	Earthquakes with a 0.35g - 0.50g peak ground acceleration
4	SEIS4	2.93E-06	2.93E-06	30.2%	12.2%	Earthquakes with a 0.50g - 1.00g peak ground acceleration
5	SEIS5	7.57E-08	7.57E-08	0.8%	0.3%	Earthquakes with a 1.00g - 1.33g peak ground acceleration
<b>Totals for All Seismic Events</b>			<b>9.70E-06</b>		<b>40.4%</b>	

The top ten seismic sequences were used to identify the dominant seismic scenarios for the current PRA models. Tables 3.B-3 (Unit 1) and 3.B-4 (Unit 2), shown below,

provide a summary for each of the Unit's top ten dominant seismic scenarios in terms of seismic initiating event, overall seismic initiator frequency, sequence core damage frequency, plant initiator (i.e., cause for the plant trip following the earthquake), and the SSCs impacted by the earthquake that are modeled in the PRA.

**Table 3.B-3**

**Unit 1 Top 10 Dominant Seismic Scenarios Impact Summary**

Rank	Seismic Initiating Event	Overall Seismic Initiator Frequency	Sequence Core Damage Frequency	Plant Initiator	PRA Modeled SSCs Impacted by Seismic Event
1	SEIS3	8.36E-06	5.67E-07	Reactor trip from loss of offsite power	Offsite grid, emergency DC power, ERF diesel generator power
2	SEIS2	1.70E-05	5.03E-07	Reactor trip from loss of offsite power	Offsite grid, emergency DC power, ERF diesel generator power
3	SEIS3	8.36E-06	4.96E-07	Reactor trip from loss of offsite power	Offsite grid, ERF diesel generator power, river water system
4	SEIS3	8.36E-06	4.90E-07	Reactor trip from loss of offsite power	Offsite grid, ERF diesel generator power, primary auxiliary building
5	SEIS2	1.70E-05	3.91E-07	Reactor trip from loss of all emergency batteries	Emergency DC power, ERF diesel generator power
6	SEIS2	1.70E-05	3.17E-07	Reactor trip from loss of offsite power	Offsite grid, ERF diesel generator power, primary auxiliary building
7	SEIS2	1.70E-05	2.82E-07	Reactor trip from loss of offsite power	Offsite grid, ERF diesel generator power, river water system
8	SEIS4	2.93E-06	2.71E-07	Reactor trip from loss of offsite power	Offsite grid, emergency DC power, ERF diesel generator power, primary auxiliary building, river water system
9	SEIS1	1.42E-04	2.63E-07	Reactor trip from loss of all emergency batteries	Emergency DC power, ERF diesel generator power
10	SEIS3	8.36E-06	2.20E-07	Reactor trip from loss of offsite power	Offsite grid, emergency DC power, ERF diesel generator power



**Table 3.B-4**

Unit 2 Top 10 Dominant Seismic Scenarios Impact Summary

Rank	Seismic Initiating Event	Overall Seismic Initiator Frequency	Sequence Core Damage Frequency	Plant Initiator	PRA Modeled SSCs Impacted by Seismic Event
1	SEIS3	8.36E-06	4.40E-07	Reactor trip from loss of offsite power	Offsite grid, normal AC/DC power, ERF diesel generator power, service water, station air compressors
2	SEIS3	8.36E-06	4.34E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, normal AC/DC power, ERF diesel generator power, station air compressors
3	SEIS3	8.36E-06	1.62E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, normal AC/DC power, ERF diesel generator power, service water, station air compressors
4	SEIS3	8.36E-06	1.59E-07	Reactor trip from loss of offsite power	Offsite grid, ERF diesel generator power, service water, station air compressors
5	SEIS3	8.36E-06	1.57E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, ERF diesel generator power, station air compressors
6	SEIS3	8.36E-06	1.44E-07	Reactor trip from loss of offsite power	Offsite grid, normal AC/DC power, emergency AC power, ERF diesel generator power, station air compressors
7	SEIS2	1.70E-05	1.14E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, normal AC/DC power, ERF diesel generator power, station air compressors
8	SEIS2	1.70E-05	1.11E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, ERF diesel generator power, station air compressors
9	SEIS3	8.36E-06	1.10E-07	Reactor trip from loss of offsite power	Offsite grid, normal AC/DC power, ERF diesel generator power, service water, station air compressors
10	SEIS3	8.36E-06	1.09E-07	Reactor trip from loss of offsite power	Offsite grid, primary auxiliary building, normal AC/DC power, ERF diesel generator power, station air compressors

However, in order to truly characterize where the seismic risk is coming from, the seismic split fraction importance of the total CDF was used. The split fraction importance shows the percent contribution to the total CDF when the seismic split fraction fails in response to the peak ground accelerations (PGA) from the seismic events. That is to say that the SSCs modeled in the seismic top events catastrophically fail at the given PGA ranges.

Tables 3.B-5 (Unit 1) and 3.B-6 (Unit 2), shown below, provide the rankings for the top ten seismic split fractions ranked in order of decreasing split fraction importance. These tables also show the split fraction value (failure probability), and the split fraction failed sequence frequency (i.e., the core damage frequency associated with sequences involving failures of the given seismic split fraction).

**Table 3.B-5**  
Unit 1 Top 10 Seismic Split Fraction Importance

Rank	Split Fraction Name	Split Fraction Importance	Split Fraction Value	Split Fraction Failed Sequence Frequency	Split Fraction Description
1	ZB3	27.0%	9.97E-01	5.28E-06	ERF diesel generator power failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
2	ZC3	25.8%	9.27E-01	5.04E-06	Offsite grid failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
3	ZB2	15.5%	9.63E-01	3.03E-06	ERF diesel generator power failures from earthquakes with a 0.25g - 0.35g peak ground acceleration
4	ZB4	15.0%	1.00E+00	2.93E-06	ERF diesel generator power failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
5	ZC4	14.9%	9.95E-01	2.92E-06	Offsite grid failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
6	ZD3	12.6%	2.94E-01	2.46E-06	Emergency DC power failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
7	ZC2	12.3%	6.77E-01	2.40E-06	Offsite grid failures from earthquakes with a 0.25g - 0.35g peak ground acceleration
8	ZG3	11.2%	2.71E-01	2.18E-06	River water system failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
9	ZP3	11.1%	2.69E-01	2.16E-06	Primary auxiliary building failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
10	ZG4	10.9%	7.23E-01	2.12E-06	River water system failures from earthquakes with a 0.50g – 1.00g peak ground acceleration

**Table 3.B-6**  
Unit 2 Top 10 Seismic Split Fraction Importance

Rank	Split Fraction Name	Split Fraction Importance	Split Fraction Value	Split Fraction Failed Sequence Frequency	Split Fraction Description
1	ZB3	18.5%	9.97E-01	4.45E-06	ERF diesel generator power failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
2	ZC3	17.4%	9.27E-01	4.18E-06	Offsite grid failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
3	ZM3	16.4%	8.83E-01	3.94E-06	Station air compressors failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
4	ZW3	13.8%	7.34E-01	3.31E-06	Normal AC/DC power failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
5	ZB4	12.2%	1.00E+00	2.93E-06	ERF diesel generator power failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
6	ZC4	12.1%	9.95E-01	2.92E-06	Offsite grid failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
7	ZM4	12.1%	9.89E-01	2.90E-06	Station air compressors failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
8	ZW4	11.2%	9.15E-01	2.68E-06	Normal AC/DC power failures from earthquakes with a 0.50g – 1.00g peak ground acceleration
9	ZG3	9.3%	2.71E-01	2.24E-06	Service water failures from earthquakes with a 0.35g - 0.50g peak ground acceleration
10	ZP3	9.2%	2.69E-01	2.22E-06	Primary auxiliary building failures from earthquakes with a 0.35g - 0.50g peak ground acceleration

In identifying potential SAMAs for seismic events, only the risk associated with peak ground acceleration ranging from 0.01g to 0.25g was used, since this range of earthquakes encompasses the site design basis earthquake value of 0.125 g. It was judged that trying to design against higher PGAs would result in excessive costs, since the structures housing the components would also need to be modified.

Tables 3.B-7 (Unit 1) and 3.B-8 (Unit 2), shown below, provide the rankings for the top five seismic split fractions associated with earthquakes ranging from 0.01g to 0.25g peak ground accelerations in order of decreasing fraction importance. As shown in Tables 3.B-7 and 3.B-8, only the top two split fractions (ZB1 and ZD1) at Unit 1 have a fraction importance greater than 1.0% (i.e., greater than a 1.0% contribution to the total CDF). Therefore, only the SSCs modeled in these split fractions' top events were considered to be significant enough to warrant a SAMA evaluation.

An examination of split fraction ZB1 (ERF diesel generator power failures from earthquakes with a 0.01g to 0.25g peak ground acceleration), reveals that 99.6% of its failure probability is dominated by the failure of the 125V DC ERF substation batteries. This seismic vulnerability was further addressed in the Unit 1 SAMA 187 evaluation.

Likewise, an investigation of split fraction ZD1 (emergency DC power failures from earthquakes with a 0.01g to 0.25g peak ground acceleration) reveals that 99.8% of its failure probability is dominated by the failure of the 125V DC battery room block walls. This seismic vulnerability was further addressed in the Unit 1 SAMA 167 evaluation.

**Table 3.B-7**

Unit 1 Top 5 SEIS1 Initiating Event Seismic Split Fraction Importance

Rank	Split Fraction Name	Fraction Importance	Risk Reduction Worth *	SF Value	Split Fraction Failed Sequence Frequency
1	ZB1	2.0%	1.016E+00	4.61E-01	4.00E-07
2	ZD1	1.9%	1.019E+00	4.94E-03	3.64E-07
3	ZC1	0.5%	1.003E+00	8.86E-02	9.03E-08
4	ZP1	0.3%	1.003E+00	1.65E-03	5.69E-08
5	ZG1	0.2%	1.002E+00	1.35E-03	4.67E-08

\* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship:  $RRW = 1 / (1 - FV)$

**Table 3.B-8**

Unit 2 Top 5 SEIS1 Initiating Event Seismic Split Fraction Importance

Rank	Split Fraction Name	Fraction Importance	Risk Reduction Worth *	SF Value	Split Fraction Failed Sequence Frequency
1	ZB1	0.5%	1.001E+00	4.61E-01	1.10E-07
2	ZP1	0.4%	1.004E+00	1.65E-03	8.59E-08
3	ZW1	0.3%	1.003E+00	1.26E-01	8.36E-08
4	ZG1	0.3%	1.003E+00	1.35E-03	7.04E-08
5	ZC1	0.2%	1.001E+00	8.86E-02	4.88E-08

\* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship:  $RRW = 1 / (1 - FV)$

#### **Question SAMA-4**

**Provide the following information concerning the MACCS2 analyses:**

- a. The MACCS2 economic input values provided in Section 3.4.2 are based on Scientech Calculation 17676-0002, "Beaver Valley Power Station MACCS2 Input Data," (Reference 33), and exceed the values provided in NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," by about 15 to 70 percent. Provide a copy of this reference.**
- b. Three problems related to use of the SECPOP2000 code have recently been identified, and publicized throughout the industry. These deal with:(1) a formatting error in the regional economic data block test file generated by SECPOP2000 for input to MACCS2 which results in MACCS2 misreading the data, (2) an error associated with the formatting of the COUNTY97.DAT economic database file used by SECPOP2000 which result in SECPOP2000 processing incorrect economic and land used data (i.e., missing entries in the "Notes" column result in data being output for the wrong county), and (3) gaps in the numbered entries in the COUNTY97.DAT economic database file which result in any county beyond county number 955 being handled incorrectly in SECPOP2000. Confirm that all three identified problems were addressed in the SAMA analyses.**

#### **RESPONSE SAMA-4.a**

- a. The MACCS2 economic input values provided in Section 3.4.2 are based on Scientech Calculation 17676-0002, "Beaver Valley Power Station MACCS2 Input Data," (Reference 33), and exceed the values provided in NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," by about 15 to 70 percent. Provide a copy of this reference.**

Enclosure A provides Scientech Calculation 17676-0002, "Beaver Valley Power Station MACCS2 Input Data." The calculation complies with the guidance of NEI 05-01.

RESPONSE SAMA-4.b

- b. Three problems related to use of the SECPOP2000 code have recently been identified, and publicized throughout the industry. These deal with:(1) a formatting error in the regional economic data block test file generated by SECPOP2000 for input to MACCS2 which results in MACCS2 misreading the data, (2) an error associated with the formatting of the COUNTY97.DAT economic database file used by SECPOP2000 which result in SECPOP2000 processing incorrect economic and land used data (i.e., missing entries in the "Notes" column result in data being output for the wrong county), and (3) gaps in the numbered entries in the COUNTY97.DAT economic database file which result in any county beyond county number 955 being handled incorrectly in SECPOP2000. Confirm that all three identified problems were addressed in the SAMA analyses.**

All three (3) of the subject SECPOP2000 code problems were identified at BVPS during SAMA preparation, communicated to the NRC and industry, and resolved with the code authors at SANDIA. The results of the revised code were checked by a group of industry experts and peers for the Beaver Valley location as well as two other licensees who were also preparing SAMA evaluations utilizing SECPOP2000 code. These code problems and resolutions were communicated to the NRC in phone calls and industry meetings. Disposition was tracked using the FENOC Corrective Action Program. The final SAMA incorporated the results of those corrective actions.



### **Question SAMA-5**

**Provide the following information with regard to the selection and screening of Phase I SAMA candidates:**

- a. Section 3.1.1.1 provides listings of the top 10 basic events based on the risk reduction worth (RRW) for CDF and large early release frequency (LERF). It also states that the basic events were identified with RRWs down to 1.005. However, the additional events (beyond the top 10) were not provided. Provide the complete listing of the basic events with RRW above the SAMA screening threshold, and identify the related SAMA candidates for each of these basic events.**
- b. Section 5.1 briefly discusses the process for SAMA identification but does not elaborate on the process details. From this brief discussion it appears that FENOC utilized both the RRW list and the direct examination of dominant sequences for the purpose of identifying the SAMA candidates. Provide a step-by-step description of the process used for SAMA identification. Include a description of the approach used to identify SAMAs that address external events. Demonstrate that potential SAMAs were considered for all dominant flood, fire, and seismic event scenarios.**
- c. Most of the potential enhancements identified in the IPE and IPEEE have been implemented or addressed by a candidate SAMA. However for a few enhancements the status is unclear. For the following enhancements, indicate if the improvement has been implemented, is no longer being considered and why, and if credit is taken for the improvement in the current PRA. For those enhancements not implemented, indicate their risk significance based on the current version of the PRA and why they should not be considered as Phase II SAMA candidates:**
  - 1. Enhance procedures and training to reduce 4.16 KV breaker failure frequencies (Units 1 and 2).**
  - 2. Locally control and align the component cooling water pumps, and locally control the B train of River Water pump during fire scenarios (Unit 1).**
- d. Reference 39, "Beaver Valley Power Station ELT 2004 Strategic Plan - Safe Plant Operations" was the source of several of potentially risk- beneficial SAMA candidates. However, this document is not discussed in the ER. Provide a copy of this document, or the portions of the document related to identification of potential plant improvements.**

- e. **Table 6-1 provides a listing of SAMA candidates, their disposition, and the associated screening criteria. However, the basis for screening some SAMA candidates was not clear. Provide additional explanation of why the following SAMAs were screened out.**
- 1. Unit 1 SAMA 73 - Proceduralize local manual operation of auxiliary feedwater system when control power is lost. Per the NRC Significance Determination Process notebook for Beaver Valley, upon loss of 480 VAC, the auxiliary feedwater valves will remain open and throttling of the valves must be performed locally. Explain why it is stated that there is no need for local manual actions. Provide an assessment of the costs and benefits of potential enhancements to improve operation of the auxiliary feedwater system when control power is lost, including improved Steam Generator (SG) level instrumentation.**
  - 2. Unit 1&2 SAMA 90 - Create a cavity flooding system. The ER indicates that the SAMA intent is met at Unit 1 using existing systems as directed by Severe Accident Management Guidelines (SAMGs), but that the SAMA was screened out at Unit 2 based on excessive cost. Explain this disparity.**

RESPONSE SAMA-5.a

- a. **Section 3.1.1.1 provides listings of the top 10 basic events based on the risk reduction worth (RRW) for CDF and large early release frequency (LERF). It also states that the basic events were identified with RRWs down to 1.005. However, the additional events (beyond the top 10) were not provided. Provide the complete listing of the basic events with RRW above the SAMA screening threshold, and identify the related SAMA candidates for each of these basic events.**

The complete list of basic events with RRW greater than or equal to 1.005 (based on internal event only) and the associated SAMAs are provided in Tables 5.A-1 and 5.A-2, shown below.

**Table 5.A-1**  
Unit 1 Basic Events Sorted by Risk Reduction Worth

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
1	HVXCRW200	1.154E+00	RW-200 MANUAL VALVE TRANSFERS CLOSED	Cooling Water SAMAs
2	CBXO480VUS18N1	1.101E+00	480V BREAKER 480VUS-1-8N1 TRANSFERS OPEN	AC Power SAMAs
3	CBXO480VUS19P1	1.096E+00	480V BREAKER 480VUS-1-9P1 TRANSFERS OPEN	AC Power SAMAs
4	XXFRACTIONRODS	1.083E+00	FRACTION OF RT FAILURES CAUSED BY CONTROL RODS FAILING TO INSERT	Anticipated Transient Without Scram (ATWS) SAMAs
5	PPRPRW3	1.076E+00	COMMON HEADER PIPE BREAK	Cooling Water SAMAs
6	FRCTRIF05	1.075E+00	FRACTION OF TIME THERE IS INSUFFICIENT RELIEF WITH 0 PORVS BLOCKED	SAMA 156
7	DGSREEEG1	1.061E+00	DIESEL GENERATOR EE-EG-1 FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
8	XXM6SS	1.059E+00	FLAG FOR MCC-1-E10 AND MCC-1-E14 AVAILABLE	N/A This event does not represent a failure
9	DGSREEEG2	1.054E+00	DIESEL GENERATOR EE-EG-2 FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
10	BSORDCSWBD2	1.049E+00	FAILURE OF 125V DC BUS 2 DC-SWBD-2 DURING 24 HR MISSION TIME	AC Power SAMAs
11	BSOR480VUS18N	1.049E+00	480V BUS 480VUS-1-8-N FAILS	AC Power SAMAs
12	BSOR4KVS1AE	1.049E+00	4160V EMERGENCY BUS 4KVS-1AE FAILS	AC Power SAMAs
13	[CBFD52BYA CBFD52BYB CBFD52RTA CBFD52RTB]	1.048E+00	REACTOR TRIP & BYPASS BREAKER DEMAND CCF	ATWS SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
14	BSOR480VUS19P	1.046E+00	480V BUS 480VUS-1-9-P FAILS	AC Power SAMAs
15	BSOR4KVS1DF	1.046E+00	4160V BUS 4KVS-1DF FAILS	AC Power SAMAs
16	CONTROLRODS	1.041E+00	CONTROL RODS FAIL TO INSERT	ATWS SAMAs
17	CBFC4KVS1A1A4	1.041E+00	SSST-1A INCOMING BKR ACB-41A (4KVS-1A-1A4) FAILS TO CLOSE	AC Power SAMAs
18	CBFC4KVS1D1D6	1.036E+00	SSST-1B INCOMING BKR ACB-341B (4KVS-1D-1D6) FAILS TO CLOSE	AC Power SAMAs
19	XXM5SS	1.036E+00	FLAG FOR MCC-1-E9 AND MCC-1-E13 AVAILABLE	N/A This event does not represent a failure
20	CBFO4KVS1A1A6	1.036E+00	USST BREAKER ACB-41C (4KVS-1A-1A6) FAILS TO OPEN	AC Power SAMAs
21	XRORTRANS18N	1.034E+00	4160V/480V TRANSFORMER TRANS-1-8N FAILS	AC Power SAMAs
22	XRORTRANS19P	1.032E+00	4160V/480V TRANSFORMER TRANS-1-9P FAILS	AC Power SAMAs
23	CBFO4KVS1D1D4	1.031E+00	USST-1D BREAKER ACB-341D (4KVS-1D-1D4) FAILS TO OPEN	AC Power SAMAs
24	[XRORTRANS18N XRORTRANS19P]	1.028E+00	4160V/480V TRANSFORMERS RUN CCF	AC Power SAMAs
25	BSORDCSWBD1	1.027E+00	FAILURE OF 125V DC BUS 1 DC-SWBD-1 DURING 24 HR MISSION TIME	DC Power SAMAs
26	[CBFC4KVS1A1A4 CBFC4KVS1D1D6]	1.026E+00	SSST BREAKERS FAILS TO CLOSE CCF	AC Power SAMAs
27	OPRXT1	1.025E+00	OPERATOR FAIL TO PERFORM CROSS TIE DURING SBO & GEN. TRANSIENT	HEP List
28	SLFDLSEE2011	1.021E+00	LEVEL SWITCH LS-EE-201-1 FAILS ON DEMAND (LOW-LOW)	ECCS SAMAs
29	OGXXXX	1.021E+00	OFFSITE GRID FAILS FOLLOWING NON-LOSP INITIATOR	LOOP SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
30	[CBFO4KVS1A1A6 CBFO4KVS1D1D4]	1.021E+00	USST BREAKERS FAILS TO OPEN CCF	AC Power SAMAs
31	OPRWA1	1.019E+00	OPERATOR FAILS TO START AUX RW PUMP GIVEN OFFSITE POWER AVAILABLE	HEP List
32	SLFDLSEE2021	1.019E+00	LEVEL SWITCH LS-EE-202-1 FAILS ON DEMAND (LOW-LOW)	ECCS SAMAs
33	[PVFRPCVRC455C]	1.019E+00	PORV PCV-RC-455C FAILS TO RECLOSE	PORV SAMAs
34	DGSSEEEG1	1.018E+00	DIESEL GENERATOR EE-EG-1 FAILS TO START	AC Power SAMAs
35	PTSRFWP2	1.017E+00	FW-P-2 FAILS TO RUN	FW/AFW SAMAs
36	[PVFRPCVRC455D]	1.017E+00	PORV PCV-RC-455D FAILS TO RECLOSE	PORV SAMAs
37	[DGSREEEG2]	1.017E+00	#2 DIESEL GENERATOR FAILS TO RUN AFTER 1ST HOUR – INDIVIDUAL CCF	AC Power SAMAs
38	FRCTRIF15	1.017E+00	FRACTION OF TIME THERE IS INSUFFICIENT RELIEF WITH 1 PORV BLOCKED	N/A This event does not represent a failure
39	[DGSREEEG1]	1.017E+00	#1 DIESEL GENERATOR FAILS TO RUN AFTER 1ST HOUR – INDIVIDUAL CCF	AC Power SAMAs
40	DGSSEEEG2	1.016E+00	DIESEL GENERATOR EE-EG-2 FAILS TO START	AC Power SAMAs
41	CBXOMCC1E10T	1.016E+00	BREAKER MCC1-E10-T TRANSFERS OPEN	AC Power SAMAs
42	[PVFRPCVRC456]	1.015E+00	PORV PCV-RC-456 FAILS TO RECLOSE	PORV SAMAs
43	XXOGSS	1.013E+00	FLAG FOR OFFSITE POWER AVAILABLE	N/A This event does not represent a failure
44	CBXO4KVS1AE1E12	1.012E+00	4160V BREAKER 4KVS-1AE-1E12 TRANSFERS OPEN	AC Power SAMAs
45	CBXO4KVS1DF1F12	1.011E+00	4160V BREAKER 4KVS-1DF-1F12 TRANSFERS OPEN	AC Power SAMAs
46	OPRRI2	1.011E+00	OPERATOR FAILS TO MANUALLY INSERT CONTROL RODS	HEP List

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
47	[XRORTRANS19P]	1.010E+00	4160V/480V TRANSFORMER TRANS-1-9P FAILS	AC Power SAMAs
48	[XRORTRANS18N]	1.010E+00	4160V/480V TRANSFORMER TRANS-1-8N FAILS	AC Power SAMAs
49	OPROC1	1.010E+00	OPERATOR FAILS TO TRIP RCP DURING LOSS OF CCR	HEP List
50	BTFDBAT1	1.009E+00	125V DC BATTERY 1 FAILS ON DEMAND	DC Power SAMAs
51	OPRCD6	1.009E+00	OPERATOR FAILS TO INITIATE COOLDOWN AND DEPRESSURIZATION (SLOCA; HH=F)	HEP List
52	CBXOMCC1E9AB	1.009E+00	BREAKER MCC1-E9-AB TRANSFERS OPEN	AC Power SAMAs
53	[SLFDLSEE2011 SLFDLSEE2021]	1.008E+00	LEVEL SWITCH FAILS ON DEMAND CCF	ECCS SAMAs
54	[PMORWRP1A PMORWRP1B]	1.008E+00	RIVER WATER PUMPS FAIL TO RUN CCF	Cooling Water SAMAs
55	BTFDBAT2	1.007E+00	125V DC BATTERY 2 FAILS ON DEMAND	DC Power SAMAs
56	OPROB1	1.007E+00	OPERATOR FAILS TO INITIATE BLEED AND FEED	HEP List
57	CBXO480VUS18N16	1.007E+00	STUB BUS BREAKER 480VUS-1-8N16 TRANSFERS OPEN	AC Power SAMAs
58	OPROF6	1.007E+00	OPERATOR FAILS TO ALIGN DEDICATED AFW AND MANUALLY CONTROL FW AFTER LOSS OF MFW AND AFW	HEP List
59	DGS1EEEG1	1.007E+00	DIESEL GENERATOR EE-EG-1 FAILS TO LOAD RUN DURING 1ST HOUR	AC Power SAMAs
60	OPRSL3	1.006E+00	OPERATOR FAILS TO GAG STUCK OPEN SAFETY RELIEF VALVE	HEP List
61	DGS1EEEG2	1.006E+00	DIESEL GENERATOR EE-EG-2 FAILS TO LOAD RUN DURING 1ST HOUR	AC Power SAMAs
62	[SLFDLSEE2021]	1.006E+00	LEVEL SWITCH LS-EE-202-1 FAILS ON DEMAND (LOW-LOW)	ECCS SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
63	BTFDBAT5	1.006E+00	125V DC BATTERY 5 FAILS ON DEMAND	DC Power SAMAs
64	[SLFDLSEE2011]	1.006E+00	LEVEL SWITCH LS-EE-201-1 FAILS ON DEMAND (LOW-LOW)	ECCS SAMAs
65	OPROB2	1.005E+00	OPERATOR FAILS TO INITIATE BLEED AND FEED	HEP List
66	[DGSSEEEG2]	1.005E+00	DIESEL GENERATOR EE-EG-2 FAILS TO START - INDIVIDUAL CCF	AC Power SAMAs
67	OPRDC1	1.005E+00	OPERATOR FAILS TO ALIGN ALTERNATE BATTERY CHARGER	HEP List
68	[DGSSEEEG1]	1.005E+00	DIESEL GENERATOR EE-EG-1 FAILS TO START - INDIVIDUAL CCF	Cooling Water SAMAs
69	[PMOSWRP1B]	1.005E+00	RIVER WATER PUMP WR-P-1B FAILS TO START	Cooling Water SAMAs
* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship: $RRW = 1 / (1 - FV)$				

**Table 5.A-2**  
Unit 2 Basic Events Sorted by Risk Reduction Worth

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
1	BSOR480VUS29	1.123E+00	BUS 480VUS-2-9 FAILS DURING OPERATION	AC Power SAMAs
2	BSOR4KVS2DF	1.123E+00	4160V BUS 4KVS-2DF FAILS DURING OPERATION	AC Power SAMAs
3	BSOR480VUS28	1.107E+00	BUS 480VUS-2-8 FAILS DURING OPERATION	AC Power SAMAs
4	BSOR4KVS2AE	1.107E+00	4160V BUS 4KVS-2AE FAILS DURING OPERATION	AC Power SAMAs
5	PTSR2FWEP22	1.096E+00	TURBINE DRIVEN PUMP 2FWE-P22 FAILS TO RUN	AFW SAMAs
6	CBFC4KVS2D2D7	1.065E+00	SSST-2B INCOMING BKR ACB-342B (4KVS-2D-2D7) FAILS TO CLOSE	AC Power SAMAs
7	CBFC4KVS2A2A4	1.059E+00	SSST-2A INCOMING BKR ACB-42A (4KVS-2A-2A4) FAILS TO CLOSE	AC Power SAMAs
8	XRORTRF29P	1.054E+00	480VUS TRANSFORMER TRF-2-9P FAILS DURING OPERATION	AC Power SAMAs
9	OGXXXX	1.050E+00	OFFSITE GRID FAILS FOLLOWING NON-LOSP INITIATOR	LOOP SAMAs
10	[FNOR2HVWFN257A FNOR2HVWFN257B FNOR2HVWFN257C]	1.047E+00	INTAKE STRUCTURE CUIBICLE HVAC FANS FAIL TO RUN CCF	Cooling Water SAMAs
11	[CBFC4KVS2A2A4 CBFC4KVS2D2D7]	1.046E+00	SSST BREAKERS FAILS TO CLOSE CCF	AC Power SAMAs
12	XRORTRF28N	1.045E+00	480V TRANSFORMER TRF-2-8N FAILS DURING OPERATION	AC Power SAMAs
13	OPRXT1	1.043E+00	OPERATOR FAIL TO PERFORM CROSS-TIE DURING SBO AND GEN. TRANSIENT	HEP List



Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
14	XXFRACTION1	1.043E+00	FRACTION OF TRANSIENTS WHEN PRESSURE RELIEF IS DEMANDED	N/A This event does not represent a failure
15	DGSR2EGSEG21	1.041E+00	DIESEL GENERATOR 2EGS-EG2-1 FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
16	DGSR2EGSEG22	1.040E+00	DIESEL GENERATOR 2EGS-EG2-2 FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
17	CBXO480VUS293B	1.037E+00	480V BREAKER 480VUS-2-9-3B TRANSFER OPEN	AC Power SAMAs
18	CBXO4KVS2DF2F11	1.037E+00	BREAKER 4KVS-2DF-2F11 TRANSFERS OPEN	AC Power SAMAs
19	CBXO480VUS283B	1.034E+00	480V BREAKER 480VUS-2-8-3B TRANSFERS OPEN	AC Power SAMAs
20	CBXO4KVS2AE2E11	1.034E+00	BREAKER 4KVS-2AE-2E11 TRANSFERS OPEN	AC Power SAMAs
21	OPROS6	1.030E+00	OPERATOR FAILS TO INITIATE AFW FOLLOWING TRANSIENT	HEP List
22	XXBPSS	1.029E+00	FLAG FOR AC PURPLE SUCCESSFUL	N/A This event does not represent a failure
23	[XRORTRF28N XRORTRF29P]	1.027E+00	480V TRANSFORMERS FAIL DURING OPERATION CCF	AC Power SAMAs
24	HVXC2QSS297	1.026E+00	MANUAL VALVE 2QSS-297 LHSI/ HHSI SUC. FROM RWST TRANSFERS CLOSED	ECCS SAMAs
25	EVFO2SVSHCV104	1.025E+00	RESIDUAL HEAT RELEASE VALVE 2SVS*HCV104 FAILS TO OPEN ON DEMAND	ECCS SAMAs
26	PVFR2RCSPCV455C	1.024E+00	PORV 2RCS-PCV455C FAILS TO RECLOSE	PORV SAMAs
27	BSORBATBKR22SWGR	1.020E+00	FAILURE OF 125V SWITCHGEAR BAT-BKR2-2-SWGR DURING 24 HRS	DC Power SAMAs
28	BSORDCSWBD22	1.020E+00	FAILURE OF 125V DC BUS DC-SWBD2-2 DURING 24 HR MISSION TIME	DC Power SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
29	RVPO2IACRV106A	1.019E+00	RECEIVER TK23 RELIEF VALVE 2IAC-RV106A OPENS PREMATURELY	Compressed Air SAMAs
30	RVPO2IACRV149	1.019E+00	RECEIVER TK21 RELIEF VALVE 2IAC-RV149 OPENS PREMATURELY	Compressed Air SAMAs
31	[CBFC4KVS2AE2E10 CBFC4KVS2DF2F10]	1.018E+00	4KV BREAKERS FAILS TO CLOSE CCF	AC Power SAMAs
32	[DGSR2EGSEG22]	1.017E+00	#2 DIESEL GENERATOR FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
33	CBFO4KVS2D2D4	1.017E+00	USST BREAKER ACB-342D (4KVS- 2D-2D4) FAILS TO OPEN	AC Power SAMAs
34	[DGSR2EGSEG21]	1.017E+00	#1 DIESEL GENERATOR FAILS TO RUN AFTER 1ST HOUR	AC Power SAMAs
35	[XRORTRF29P]	1.016E+00	480VUS TRANSFORMER TRF-2-9P FAILS DURING OPERATION	AC Power SAMAs
36	[XRORTRF28N]	1.016E+00	480V TRANSFORMER TRF-2-8N FAILS DURING OPERATION	AC Power SAMAs
37	CBFO4KVS2A2A7	1.016E+00	USST BREAKER ACB-42C (4KVS- 2A-2A7) FAILS TO OPEN	AC Power SAMAs
38	[CBFO4KVS2A2A7 CBFO4KVS2D2D4]	1.013E+00	USST BREAKERS FAIL TO OPEN CCF	AC Power SAMAs
39	FRCTRIF05	1.013E+00	FRACTION OF TIME THERE IS INSUFFICIENT RELIEF WITH 0 PORVS BLOCKED	SAMA 156
40	CBFC4KVS2AE2E10	1.012E+00	BREAKER 4KVS-2AE-2E10 FAILS TO CLOSE	AC Power SAMAs
41	CBFC4KVS2DF2F10	1.012E+00	BREAKER 4KVS-2DF-2F10 FAILS TO CLOSE	AC Power SAMAs
42	OPROF2	1.012E+00	OPERATOR FAILS TO REALIGN MAIN FEEDWATER - NO SI SIGNAL	HEP List
43	XXFRACTIONRODS	1.011E+00	FRACTION OF RT FAILURES CAUSED BY CONTROL RODS FAILING TO INSERT	ATWS SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
44	PTSS2FWEP22	1.011E+00	TURBINE DRIVEN PUMP 2FWE-P22 FAILS TO START	AFW SAMAs
45	PVFR2RCSPCV455D	1.009E+00	PORV 2RCS-PCV455D FAILS TO RECLOSE	PORV SAMAs
46	PVFR2RCSPCV456	1.009E+00	PORV 2RCS-PCV456 FAILS TO RECLOSE	PORV SAMAs
47	BTFDBAT22	1.009E+00	125V DC BATTERY BAT-2-2 FAILS ON DEMAND	DC Power SAMAs
48	BTFDBAT26	1.008E+00	BATTERY BAT-2-6 FAILS ON DEMAND AFTER UNIT TRIP	DC Power SAMAs
49	[CBFD52BYA CBFD52BYB CBFD52RTA CBFD52RTB]	1.008E+00	REACTOR TRIP & BYPASS BREAKERS FAIL ON DEMAND CCF	ATWS SAMAs
50	BTFDBAT21	1.008E+00	125V DC BATTERY BAT-2-1 FAILS ON DEMAND	DC Power SAMAs
51	[PMSS2FWEP23A]	1.008E+00	MD AFW PUMP 2FWE-P23A FAILS TO START	AFW SAMAs
52	[EVFO2SVSPCV101C]	1.008E+00	ATMOSPHERIC RELIEF VALVE 2SVS*PCV101C FAILS TO OPEN ON DEMAND	Depressurization SAMAs
53	[EVFO2SVSPCV101A]	1.007E+00	ATMOSPHERIC RELIEF VALVE 2SVS*PCV101A FAILS TO OPEN ON DEMAND	Depressurization SAMAs
54	BTFDBAT25	1.007E+00	BATTERY BAT-2-5 FAILS ON DEMAND AFTER UNIT TRIP	DC Power SAMAs
55	[EVFO2SVSPCV101B]	1.007E+00	ATMOSPHERIC RELIEF VALVE 2SVS*PCV101B FAILS TO OPEN ON DEMAND	Depressurization SAMAs
56	DGS12EGSEG21	1.007E+00	DIESEL GENERATOR 2EGS-EG2-1 FAILS TO LOAD RUN DURING 1ST HOUR	AC Power SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
57	XXAMSACFAILS	1.007E+00	AMSAC FAILS TO GENERATE SIGNAL	ATWS SAMAs
58	DGS12EGSEG22	1.007E+00	DIESEL GENERATOR 2EGS-EG2-2 FAILS TO LOAD RUN DURING 1ST HOUR	AC Power SAMAs
59	MVFO2SWSMOV113A	1.007E+00	SWS TO 2EGS-DG2-1 ISOLATION MOV 2SWS-MOV113A FAILED TO OPEN	Cooling Water SAMAs
60	MVFO2SWSMOV113D	1.007E+00	SWS TO 2EGS-DG2-2 ISOLATION MOV 2SWS-MOV113D FAILED TO OPEN	Cooling Water SAMAs
61	LOFDWR	1.007E+00	FEEDWATER FAILURE DURING MISSION TIME	AFW SAMAs
62	BSORBATBKR21SWGR	1.006E+00	FAILURE OF 125V SWITCHGEAR BAT-BKR2-1-SWGR DURING 24 HRS	DC Power SAMAs
63	BSORDCSWBD21	1.006E+00	FAILURE OF 125V DC BUS DC-SWBD2-1 DURING 24 HR MISSION TIME	DC Power SAMAs
64	CRFD2RPS001	1.006E+00	CONTROL RODS FAIL TO INSERT	ATWS SAMAs
65	[PMOR2SWSP21A PMOR2SWSP21B PMOR2SWSP21C]	1.006E+00	SERVICE WATER PUMPS FAIL TO RUN CCF	Cooling Water SAMAs
66	DGSS2EGSEG21	1.006E+00	DIESEL GENERATOR 2EGS-EG2-1 FAILS TO START	AC Power SAMAs
67	[DGSR2EGSEG21 DGSR2EGSEG22]	1.006E+00	DIESEL GENERATOR FAIL TO RUN AFTER 1ST HOUR CCF	AC Power SAMAs
68	[FNOR2HVWFN257B]	1.006E+00	C-CUIBICLE VENTIL. FAN 2HVW-FN257B FAILS TO RUN	Cooling Water SAMAs
69	DGSS2EGSEG22	1.006E+00	DIESEL GENERATOR 2EGS-EG2-2 FAILS TO START	AC Power SAMAs
70	[MVFO2SWSMOV113A MVFO2SWSMOV113D]	1.006E+00	SWS TO EDG ISOLATION MOVS FAIL TO OPEN CCF	Cooling Water SAMAs

Rank	Basic Event	BE Risk Reduction Worth *	Basic Event Description	Associated SAMA
71	PMSR2FWEP23A1	1.006E+00	LUBE OIL PUMP 2FWE-P23A1 FAILS TO RUN	AFW SAMAs
72	[PMSR2FWEP23A]	1.005E+00	MD AFW PUMP 2FWE-P23A FAILS TO RUN	AFW SAMAs
73	OPRCD6	1.005E+00	OPERATOR FAILS TO COOLDOWN AND DEPRESSURIZE (SLOCA AND HH=F)	HEP List
74	[PMOR2SWSP21B]	1.005E+00	SERVICE WATER PUMP 2SWS-P21B FAILS TO RUN	Cooling Water SAMAs
75	[CBFO4KVS2AE2E7 CBFO4KVS2DF2F7]	1.005E+00	NORMAL SUPPLY BREAKERS FAIL TO OPEN CCF	AC Power SAMAs
76	OPRSL1	1.005E+00	OPERATOR FAILS TO IDENTIFY RUPTURED S-G OR INITIATE ISOLATION	HEP List
77	CVFR2SIS27	1.005E+00	CHECK VALVE 2SIS-27 FAILS TO RESEAT	ECCS SAMAs
* The Risk Reduction Worth (RRW) is defined by the following Fussell-Vesley (FV) relationship: $RRW = 1 / (1 - FV)$				

RESPONSE SAMA-5.b

- b. Section 5.1 briefly discusses the process for SAMA identification but does not elaborate on the process details. From this brief discussion it appears that FENOC utilized both the RRW list and the direct examination of dominant sequences for the purpose of identifying the SAMA candidates. Provide a step-by-step description of the process used for SAMA identification. Include a description of the approach used to identify SAMAs that address external events. Demonstrate that potential SAMAs were considered for all dominant flood, fire, and seismic event scenarios.**

The SAMA analysis consisted of the following elements:

1. Include as SAMA candidates all the items identified in Table 14 of NEI 05-01, to capture the generic industry list of potential improvements for PWR;
2. Review the Individual Plant Examination (IPE) and the Individual Plant Examination of External Events (IPEEE), and identify the potential enhancements that are suggested in these analyses;
3. Identify the potential IPE/IPEEE enhancements that have not already been implemented and designate these as potential SAMA candidates and include them on the SAMA list;
4. Obtain the risk reduction worth listing of basic events from the current PRA for those events having  $RRW > 1.005$ ;
5. Obtain the risk reduction worth listing of systems from the current PRA for those systems having  $RRW > 1.005$ ;
6. Perform a comparison of the basic events with  $RRW > 1.005$  with the objective of identifying potential SAMA candidates;
7. Perform a comparison of the system listing of systems with  $RRW > 1.005$  with the SAMA list to assure that each system relates to a SAMA candidate, adding candidates for those that do not correlate to one of the existing candidates;
8. Perform a comparison of the SAMA candidates with the top sequences to determine that each of the sequences includes at least one contributor that is addressed by one of the SAMA candidates, adding candidates for those that do not correlate to one of the existing candidates. Sequences that contribute more than 1% to total core damage were considered; and,
9. Interviews with plant personnel and review of other plant documents identified additional SAMA candidates.

The external events are modeled directly in the PRA and were included in the lists described above. The analysis of internal and external events was performed

simultaneously. The top 10 initiator contribution and sequences are listed in Tables 5.B-1, 5.B-2, 5.B-3, and 5.B-4, shown below.

The response to RAIs SAMA-1.h for flood, SAMA-3.a for fire, and SAMA-3.b for seismic events provide potential SAMAs that were considered for these dominant scenarios.

While preparing the response for this RAI, it was determined that fire initiator CR-1 (control room fire) was omitted from the SAMA evaluation. Analysis of this Unit 1 SAMA (denoted as SAMA 190) has subsequently been performed. See Enclosure B for ER Unit 1 SAMA 190 information.

**Table 5.B-1**  
Unit 1 Top 10 Individual CDF Initiating Events

Rank	Initiating Event	Core Damage Frequency (per yr.)	Percent CDF Contribution	Associated SAMAs
1	Earthquakes (0.35g to 0.50g)	5.29E-06	27.1%	Seismic SAMAs
2	Earthquakes (0.25g to 0.35g)	3.09E-06	15.8%	Seismic SAMAs
3	Earthquakes (0.5g to 1.0g)	2.93E-06	15.0%	Seismic SAMAs
4	Loss of an Emergency AC Power Train	1.31E-06	6.7%	AC Power SAMAs
5	Cable Fire in CV-3	6.30E-07	3.2%	SAMA 180
6	Earthquakes (0.10 to 0.25g)	5.04E-07	2.6%	Seismic SAMAs
7	Cable Fire in CR-1	4.59E-07	2.3%	SAMA 190 (See Enclosure B)
8	Cable Fire in CS-1	3.61E-07	1.8%	SAMA 183
9	HVAC Fire in CS-1	2.81E-07	1.4%	SAMA 181
10	Loss of All River Water	2.67E-07	1.4%	Cooling Water SAMAs
	All Others	4.42E-06	22.6%	
	<b>All Initiating Events</b>	<b>1.95E-05</b>	<b>100%</b>	



**Table 5.B-2**  
Unit 1 Top 10 Sequences Contributing to Core Damage

Rank No.	Initiating Event	Frequency (per year)	Percent of CDF	Associated SAMA
1	Earthquake Between 0.35 – 0.50 g's	5.67E-07	2.9%	Seismic SAMAs
2	Cable Fire in CV (Cable Tunnel)	5.60E-07	2.9%	SAMA 180
3	Earthquake Between 0.25 – 0.35 g's	5.03E-07	2.6%	Seismic SAMAs
4	Earthquake Between 0.35 – 0.50 g's	4.96E-07	2.5%	Seismic SAMAs
5	Earthquake Between 0.35 – 0.50 g's	4.90E-07	2.5%	Seismic SAMAs
6	Cable Fire in CR-1 (Control Room General Area)	4.11E-07	2.1%	SAMA 190 (See Enclosure B)
7	Earthquake Between 0.25 – 0.35 g's	3.91E-07	2.0%	Seismic SAMAs
8	Cable Fire in CS-1 (Cable Spreading Room)	3.21E-07	1.6%	SAMA 183
9	Earthquake Between 0.25 – 0.35 g's	3.17E-07	1.6%	Seismic SAMAs
10	Earthquake Between 0.25 – 0.35 g's	2.82E-07	1.4%	Seismic SAMAs

**Table 5.B-3**  
Unit 2 Top 10 Individual CDF Initiating Events

Rank	Initiating Event	Core Damage Frequency (per yr.)	Percent CDF Contribution	Associated SAMAs
1	Earthquakes (0.35g to 0.50g)	4.46E-06	18.5%	Seismic SAMAs
2	Loss of an Emergency AC Power Train	3.80E-06	15.8%	AC Power SAMAs
3	Earthquakes (0.5g to 1.0g)	2.93E-06	12.2%	Seismic SAMAs
4	Earthquakes (0.25g to 0.35g)	2.02E-06	8.4%	Seismic SAMAs
5	EDG Building Fires DG1L1A / DG2L1A	1.82E-06	7.6%	SAMA 184/185
6	Loss of Offsite Power Extreme Weather	6.61E-07	2.7%	LOOP SAMAs
7	Cable Vault Flood from Fire Water	6.07E-07	2.5%	SAMA 187
8	Total Loss of Service Water	5.29E-07	2.2%	Cooling Water SAMAs
9	Loss of an Emergency DC Power Train	5.18E-07	2.2%	DC Power SAMAs
10	Cable Tunnel Fire CT1L1A	6.19E-06	2.1%	SAMA 180
-	All Other	6.19E-06	25.7%	
	<b>ALL INITIATING EVENTS</b>	<b>2.40E-05</b>	<b>100%</b>	

**Table 5.B-4**  
Unit 2 Top 10 Sequences Contributing to Core Damage

<b>Rank No.</b>	<b>Initiating Event</b>	<b>Frequency (per year)</b>	<b>Percent of CDF</b>	<b>Associated SAMAs</b>
1	Loss of Emergency 4160V AC Orange Power with failure to establish AC power crosstie	6.68E-07	2.8%	AC Power SAMAs
2	Loss of Emergency 4160 V AC Purple Power with failure to establish AC power crosstie	6.55E-07	2.7%	AC Power SAMAs
3	Total Loss of Service Water Trains A & B	4.70E-07	2.0%	Cooling Water SAMAs
4	Earthquake (.35g to .50g) with Primary Intake Building Failure	4.40E-07	1.8%	Seismic SAMAs
5	Earthquake (.35g to .50g) with Primary Auxiliary Building Failure	4.34E-07	1.8%	Seismic SAMAs
6	Cable Fire in Cable Tunnel CT-1	3.53E-07	1.5%	SAMA 180
7	#2 EDG Fire DG-2	3.52E-07	1.5%	SAMA 185
8	#1 EDG Fire DG-1	3.52E-07	1.5%	SAMA 184
9	Loss of Emergency 4160V AC Orange Power with failure of offsite power infeed breaker	2.80E-07	1.2%	AC Power SAMAs
10	Loss of Emergency 4160 V AC Purple Power with failure of offsite power infeed breaker	2.77E-07	1.2%	AC Power SAMAs

RESPONSE SAMA-5.c.1

- c. Most of the potential enhancements identified in the IPE and IPEEE have been implemented or addressed by a candidate SAMA. However for a few enhancements the status is unclear. For the following enhancements, indicate if the improvement has been implemented, is no longer being considered and why, and if credit is taken for the improvement in the current PRA. For those enhancements not implemented, indicate their risk significance based on the current version of the PRA and why they should not be considered as Phase II SAMA candidates:**

**1. Enhance procedures and training to reduce 4.16 KV breaker failure frequencies (Units 1 and 2).**

Unit 1 and Unit 2 Tables 5.2-1 in the ER, Attachments C-1 and C-2, pages C.1-64 and C.2-65, identified the Beaver Valley IPE vulnerabilities and potential enhancements. As shown on these tables, the fast 4.16 KV bus transfer failures, which are associated with the 4.16 KV breaker failures, were identified as vulnerabilities at each Unit. The potential enhancements for these failures through explicit procedure and training on breaker repair or change out were identified as "intent met" by Unit 1 SAMA 161 and Unit 2 SAMA 21.

The basis for stating that the intent was met is that both procedures and training for manually racking 4KV breakers have been enhanced, and spare breaker internals are available near the required locations to support the replacement. Therefore, this improvement has been implemented and is no longer considered. The current procedure provides thorough, step-by-step instructions for racking 4.16 KV breakers, complete with breaker diagrams and caution notes.

In addition, 4.16 KV breaker racking is taught as part of operator training. It is taught in initial non-licensed operator training, and as on-the-job training in the plant during license class. While in training, operators practice on 4.16 KV breakers in the Maintenance Training Center. They are also shown a video that explains possible mistakes that can be made when racking a breaker, as well as indications of properly and improperly racked breakers. Additionally, 4.16 KV breaker racking is covered periodically in operator requalification training.

The operator action to recover a failure of the 4.16 KV fast bus transfer breakers is currently modeled in both Unit's PRA models; specifically in both Unit's electric power recovery top event as operator actions ZHERE5, ZHERE6, ZHERED, and ZHEREE, and the AC power crosstie top event as operator action OPRXT4. These actions give credit for operator actions to identify, replace, rack-in, and energize a failed fast bus transfer 4.16 KV breaker. The existing revision of the procedure for racking 4.16 KV breakers and the operator training knowledge, as well as the specific procedures identifying the need for the action were factored in the human reliability analysis for

determining the human error rates of the operator actions. Therefore, credit has been taken for this improvement in the current PRA models.

#### RESPONSE SAMA-5.c.2

**c. Most of the potential enhancements identified in the IPE and IPEEE have been implemented or addressed by a candidate SAMA. However for a few enhancements the status is unclear. For the following enhancements, indicate if the improvement has been implemented, is no longer being considered and why, and if credit is taken for the improvement in the current PRA. For those enhancements not implemented, indicate their risk significance based on the current version of the PRA and why they should not be considered as Phase II SAMA candidates:**

- 2. Locally control and align the component cooling water pumps, and locally control the B train of River Water pump during fire scenarios (Unit 1).**

The Beaver Valley IPEEE analysis identified three fire areas (PA-1E, CS-1, and NS-1) where operator actions could be enhanced for locally controlling and aligning component cooling water pumps (or locally opening RCP seal injection MOVs), or locally controlling the B Train River Water pump. The Fire Emergency Procedures (FEPs) currently include specific guidance for locally starting the B Train River Water Pump by closing the supply breaker for a fire in area CS-1 or NS-1. Based on these procedures, improvement opportunities have been implemented for the CS-1 and NS-1 fire areas, and no new enhancements are being considered at this time.

The fire scenarios modeled in the BV1REV4 fire PRA are those created for the original IPEEE analysis and have not been updated to include any enhancements, procedural or otherwise. The IPEEE fire PRA credited these local operator actions based on the procedures in effect at the time to reduce the fire initiating event frequency by some non-recovery factors (i.e., human error probability for performing these actions), which are still used in the BV1REV4 fire PRA model. Therefore, the current Unit 1 PRA model (BV1REV4) does not take credit for any of these procedure improvements.

For a fire originating in the PA-1E fire area, there is currently no explicit FEP guidance for locally controlling and aligning component cooling water pumps or locally opening RCP seal injection MOVs, but the IPEEE did credit alarm response procedures for performing these actions. However, based on the current version of the Unit 1 PRA model (BV1REV4), the summation of the PA-1 fire scenarios has less than a 0.1% contribution to the total core damage frequency. Due to the low CDF contribution, PA-1E fires no longer warrant any further SAMA considerations.

Further enhancements regarding these identified opportunities are not currently being considered; however, FENOC has made the decision (see FENOC letter to NRC dated December 22, 2005, (ADAMS accession number ML060040259)) to transition to the new NFPA 805 fire protection program in accordance with 10 CFR 50.48(c). An entirely new Fire PRA will need to be developed to support this transition and would then be used to identify, and subsequently address, risk-significant areas for improvement.

RESPONSE SAMA-5.d

- d. Reference 39, "Beaver Valley Power Station ELT 2004 Strategic Plan - Safe Plant Operations" was the source of several of potentially risk- beneficial SAMA candidates. However, this document is not discussed in the ER. Provide a copy of this document, or the portions of the document related to identification of potential plant improvements.**

Reference 39, "Beaver Valley Power Station ELT 2004 Strategic Plan – Safe Plant Operations," was referenced in the ER as a source document since it represented a historical perspective based on the first BVPS LRA submitted in February 2005, and provided an initial starting point for the development of BVPS SAMAs for the application currently under review. The "Conclusions" section of the plan identifies the potential plant improvements that were considered by BVPS staff members in 2004 based on the previous BV1REV3 and BV2REV3B PRA models; the relevant information from this section has been extracted and provided below. It should be noted that the improvements listed in the 2004 Strategic Plan were considered during the recent BVPS SAMA process, and the specific SAMA numbers (bold-italics) from the BVPS ER have been annotated in parentheses in the 2004 Strategic Plan excerpted tables, "Unit 1 Potential Safety Margin Improvements," and "Unit 2 Potential Safety Margin Improvements," shown below.

-----BEGIN EXCERPT OF 2004 STRATEGIC PLAN-----

**ELT Strategic Action Plan – Safe Plant Operations  
MII. Margin Improvements”**

Objective:

Evaluate Beaver Valley safety margin improvement opportunities using PSA insight and present the most significant opportunities to the Senior and Executive Leadership Teams for consideration.

Conclusions:

BVPS Unit 1

At Beaver Valley Unit 1 the current PRA model Core Damage Frequency (CDF) due to internal and external initiating events is  $2.34 \times 10^{-5}$  per year, and the associated Large Early Release Frequency (LERF) is  $9.99 \times 10^{-7}$  per year. Major reductions in CDF or LERF can be achieved through the implementation of a few modifications to the plant and procedures. The following table presents the top five safety margin improvement opportunities at BVPS Unit 1 presented in the order of most beneficial of reducing the risk at BVPS-1, assuming that the modification completely eliminates its contribution to risk. These are based on a draft copy of the Severe Accident Mitigation Alternatives (SAMA) performed for the BVPS License Renewal Application with some modifications.

**Table: Unit 1 Potential Safety Margin Improvements**

Rank	Potential Safety Margin Improvement Description	Maximum Reduction in CDF or LERF (per year)
1	Install an Independent RCP Seal Injection System ( <i>SAMA 165</i> )	19% (CDF)
2	Increase the Seismic Ruggedness of the Emergency 125V DC Battery Block Walls ( <i>SAMA 167</i> )	13% (CDF)
3	Install Fire Barriers for HVAC Fans in the Cable Spreading Room (CS-1) ( <i>SAMA 168</i> )	13% (CDF)
4	Increase Reliability of Emergency 125V DC Busses ( <i>SAMA 163</i> )	4% (CDF)
5	Revise Emergency Procedures to Isolate a Faulted SG due to Stuck-Open Safety Valve ( <i>SAMA 164</i> )	56% (LERF)

BVPS Unit 2

At Beaver Valley Unit 2 the current PRA model Core Damage Frequency due to internal and external initiating events is  $3.43 \times 10^{-5}$  per year, and the associated Large Early Release Frequency is  $1.14 \times 10^{-6}$  per year. Major reductions in CDF or LERF can be achieved through the implementation of a few modifications to the plant and procedures. The following table presents the top five safety margin improvement opportunities at BVPS Unit 2 presented in the order of most beneficial of reducing the risk at BVPS-2, assuming that the modification completely eliminates its contribution to risk. These are based on a draft copy of the Severe Accident Mitigation Alternatives (SAMA) performed for the BVPS License Renewal Application with some modifications.

**Table: Unit 2 Potential Safety Margin Improvements**

Rank	Potential Safety Margin Improvement Description	Maximum Reduction in CDF or LERF (per year)
1	Provide Additional Emergency 125V DC Battery Capability ( <i>SAMA 166</i> )	38% (CDF)
2	Install an Independent RCP Seal Injection System ( <i>SAMA 165</i> )	28% (CDF)
3	Improve SGTR Coping Capability-Include Diesel Driven FW Pump ( <i>SAMA 162</i> )	12% (CDF)
4	Increase Reliability of Emergency 125V DC Busses ( <i>SAMA 163</i> )	6% (CDF)
5	Revise Emergency Procedures to Isolate a Faulted SG due to Stuck-Open Safety Valve ( <i>SAMA 164</i> )	46% (LERF)

-----END EXCERPT OF 2004 STRATEGIC PLAN-----



RESPONSE SAMA-5.e.1

**e. Table 6-1 provides a listing of SAMA candidates, their disposition, and the associated screening criteria. However, the basis for screening some SAMA candidates was not clear. Provide additional explanation of why the following SAMAs were screened out.**

- 1. Unit 1 SAMA 73 - Proceduralize local manual operation of auxiliary feedwater system when control power is lost. Per the NRC Significance Determination Process notebook for Beaver Valley, upon loss of 480 VAC, the auxiliary feedwater valves will remain open and throttling of the valves must be performed locally. Explain why it is stated that there is no need for local manual actions. Provide an assessment of the costs and benefits of potential enhancements to improve operation of the auxiliary feedwater system when control power is lost, including improved Steam Generator (SG) level instrumentation.**

The second sentence in Unit 1 Table 6-1 "Phase I Disposition" column for SAMA 73 incorrectly states, "During an SBO, no manual actions are needed for TDAFW operation," when, in fact, procedures do exist for local manual equipment operation. The Table 6-1 "Phase I Disposition" column entry for Unit 1 SAMA 73 should read in its entirety, "Already Implemented. Procedure exists." This revised entry would then match the one contained in Unit 2 Table 6-1 for the identical Unit 2 SAMA 73. Cost and benefit analysis is not required since this SAMA is already implemented.

See Enclosure B for ER revisions to Unit 1 SAMA 73 information.

RESPONSE SAMA-5.e.2

**e. Table 6-1 provides a listing of SAMA candidates, their disposition, and the associated screening criteria. However, the basis for screening some SAMA candidates was not clear. Provide additional explanation of why the following SAMAs were screened out.**

- 2. Unit 1&2 SAMA 90 - Create a cavity flooding system. The ER indicates that the SAMA intent is met at Unit 1 using existing systems as directed by Severe Accident Management Guidelines (SAMGs), but that the SAMA was screened out at Unit 2 based on excessive cost. Explain this disparity.**

The Unit 2 BV SAMA 90 was incorrectly screened. SAMA 90 screens out for both units, and the rationale for screening should be the same for both units. In both units, the cost associated with creating a new system would be prohibitively high. However, both unit SAMGs provide guidance for the injection of multiple RWST volumes into the

containment to flood the cavity in efforts to prevent or mitigate the consequences associated with core-concrete interactions. Furthermore, both Unit's containments have holes in the reactor cavity wall to allow water in the cavity to drain out to the sumps, but these holes would also allow water to flow into the cavity if multiple RWST volumes are injected inside the containment.

The intent of SAMA 90 is, therefore, met for both units.

See Enclosure B for ER revisions to Unit 2 SAMA 90 information.

### **Question SAMA-6**

**Provide the following information with regard to the Phase II cost-benefit evaluations:**

- a. SAMA 41 involves primary system depressurization and use of low pressure injection (LPI) when high pressure injection (HPI) has failed. However, it appears to have already been credited in the latest PRA revision. Describe the additional modifications and enhancements that are included in the scope of this SAMA and discuss the basis for the cost estimate.**
- b. SAMAs 55, 56, and 165 were considered as alternative approaches for reducing the likelihood of reactor coolant pump (RCP) seal LOCAs. SAMA 55 would eliminate seal LOCAs for all initiators, whereas SAMAs 56 and 165 would eliminate seal LOCAs for all initiators except SBO. The different scopes of these SAMAs do not appear to have been considered in the cost-benefit evaluation since the same cost and benefit values were used for all three SAMAs. Provide either separate cost and benefit estimates for each of these SAMAs, or confirmation that the implementation cost and benefit estimates used to represent these SAMAs are bounding (i.e., the implementation cost is the lowest value and the benefit estimate is the highest value of the three SAMAs).**
- c. SAMA 98 involves increasing the containment and core debris cooling following core damage, however, the benefit is estimated based on eliminating containment failures from hydrogen burn. Explain why the benefit of this SAMA would be equivalent to that associated with eliminating hydrogen burns. Also explain why the cost estimate for this SAMA is lower for Unit 1 than for Unit 2.**
- d. For SAMAs 112 and 113 the implementation costs appear to be higher than expected. Provide the basis for the cost estimates for these SAMAs.**
- e. The benefit of implementing a SAMA is estimated by summing the reductions in four major severe accident costs: offsite exposure cost, off-site economic cost, onsite exposure cost, and on-site economic cost. Table 7-1 provides the reduction in offsite dose and CDF but does not include the reduction in Offsite Economic Cost Risk (OECR). Provide the reduction in OECR for each SAMA.**

RESPONSE SAMA-6.a

- a. **SAMA 41 involves primary system depressurization and use of low pressure injection (LPI) when high pressure injection (HPI) has failed. However, it appears to have already been credited in the latest PRA revision. Describe the additional modifications and enhancements that are included in the scope of this SAMA and discuss the basis for the cost estimate.**

As listed in the Table 7-1 (for each unit), SAMA 41 involves creation of a reactor coolant depressurization system which would allow low pressure Emergency Core Cooling System injection in the event of a small LOCA in conjunction with a high pressure safety injection failure. The individual potential benefits for each unit are estimated to be \$48.0K at Unit 1 and \$83.8K at Unit 2. The current plant designs and PRA models already include design features controlled by plant procedures (ref. SAMA 42), which account for loss of high pressure safety injection during a small-break LOCA.

The existing design features utilized during this scenario include: opening the atmospheric steam dump valves (ASDVs) on each steam generator (for rapid Reactor Coolant System (RCS) depressurization); thereby, reducing RCS pressure to below the safety injection accumulator set pressure (which would passively begin injecting into the RCS and reflood the core). Also, the pressurizer power-operated relief valves (PORVs) could be opened, if needed, to assist in further depressurization of the RCS to the point where low pressure safety injection can begin.

An Expert Panel (EP) was used to assess the value (cost-benefit) of proposed SAMA plant changes (equipment, procedures, processes) to identify beneficial candidates for inclusion in the LRA. The EP focused primarily on estimating the cost of SAMA implementation to perform the cost-benefit comparison; benefit values having been previously established.

The EP process relied upon the expertise and judgment of eight (8) long-term site staff members drawn from the engineering, operations, and training departments, and four (4) support contractors (SAMA & PRA). Several of the site staff also had previous assignments in maintenance and quality assurance. The EP collectively understood the plant's configuration and operation – including unit differences, as well as the plant's design, licensing, and training requirements and processes. Consequently, the EP was able to identify the array of plant changes that would be both required for – and result from, SAMA implementation.

The EP then subjectively approximated the cost of performing all associated implementation items—design changes, material requirements, installation work, procedure changes, training, regulatory effort, and long-term maintenance. Some costs were assigned a standard value based upon past experience and estimated man-hours required (e.g., procedure development = \$15K; minimal physical plant change = \$100K). Least cost “out-of-the-box” options were included wherever possible (e.g.,

securing retail store small generator(s)). Detailed design concepts were not developed by the EP, but every effort was made to identify and reasonably price all activities that needed to be performed in support of each SAMA candidate (i.e., “conceptually estimated”, NEI 05-01, Section 7.2, “Cost of SAMA Implementation”). Consensus on a final estimated cost was reached, but not all cost components were documented. In all cases, however, the EP attempted to consider credible cost data offered by the members, and to establish a rationale for including – not excluding, SAMA candidates in the final recommended implementation group. No SAMAs were excluded because an aging effect did not exist.

When appropriate, the EP members also used knowledge of plant configuration and operation, and previous improvements, to determine whether a proposed SAMA’s “intent” had already been met.

The Expert Panel estimated the cost of SAMA 41 for the installation of a new independent reactor coolant depressurization system would exceed \$1,000K per unit, with the consideration that current plant design features exist at both units that already meet the intent of SAMA 41. The estimated cost at each unit was based in a conceptual design involving installation of safety-related piping, supports, locally and remotely operated valve(s), and power & control circuits necessary to achieve a controlled depressurization of the RCS.

Five (5) meetings of the EP were held to reach a consensus on the recommended SAMA candidates.

#### RESPONSE SAMA-6.b

**b. SAMAs 55, 56, and 165 were considered as alternative approaches for reducing the likelihood of reactor coolant pump (RCP) seal LOCAs. SAMA 55 would eliminate seal LOCAs for all initiators, whereas SAMAs 56 and 165 would eliminate seal LOCAs for all initiators except SBO. The different scopes of these SAMAs do not appear to have been considered in the cost-benefit evaluation since the same cost and benefit values were used for all three SAMAs. Provide either separate cost and benefit estimates for each of these SAMAs, or confirmation that the implementation cost and benefit estimates used to represent these SAMAs are bounding (i.e., the implementation cost is the lowest value and the benefit estimate is the highest value of the three SAMAs).**

FENOC confirms that the different SAMA scopes were considered in developing the bounding values of implementation cost and benefit estimates used for the three SAMAs in question. The assessment approach taken with this group of SAMAs was similar to the approach taken for other related SAMAs. This process results in bounding implementation cost and benefit values being applied to similar SAMAs. However, the

cost value for the least costly alternative was compared with the maximum benefit achievable from any of these three alternatives to consistently provide a bounding assessment of the three SAMAs.

The SAMA case used to evaluate these 3 SAMAs was RCPLOCA2. This case determines the benefit of eliminating all RCP seal LOCA events except those associated with seismic events with a peak ground acceleration PGA greater than 0.35g. This allows evaluation of various possible improvements that could reduce the risk associated with RCP seal LOCA and other small LOCA events. This case would estimate the maximum benefit that could be attained by improvements that would reduce the likelihood of RCP seal LOCA other than those failures caused by severe mechanical shaking of high PGA earthquakes (and there is no reason to assume that reasonably designed alternative systems would be exempt from such damage also). This maximum benefit (\$1358K) has then been compared with the costs of implementation of the SAMAs.

The costs of implementing these SAMAs have been individually considered by the Expert Panel as described above. The cost of installing new systems with the attendant need to install new piping, possibly new containment penetrations, and possibly, for independence, new power supplies, and including the maintenance issues associated with the systems and new penetrations, is extremely high over the extended life. These costs were not developed in detail because the Expert Panel knew from extensive experience with plant modifications that the costs would be well in excess of \$4000K for even the least costly of these SAMAs. Therefore, the least cost was applied to all three SAMAs. Since the least cost far exceeded the maximum benefit that could be achieved by implementation of any of the three SAMAs, they all screened out. The differences in scope of the three SAMAs were therefore considered, even though the values actually used for screening were the same bounding values.

#### RESPONSE SAMA-6.c

- c. SAMA 98 involves increasing the containment and core debris cooling following core damage, however, the benefit is estimated based on eliminating containment failures from hydrogen burn. Explain why the benefit of this SAMA would be equivalent to that associated with eliminating hydrogen burns. Also explain why the cost estimate for this SAMA is lower for Unit 1 than for Unit 2.**

SAMA 98, "Create a core melt source reduction system," was considered to be focused on minimizing interaction of core debris and containment concrete (i.e., core-concrete interaction). The core-concrete interaction between the corium with the concrete beneath the reactor vessel following a high pressure melt ejection produces significant hydrogen that presents a risk of combustion and potential containment failure. The installation of refractory material placed underneath the reactor vessel would minimize

or prevent any core-concrete interactions, thereby reducing the hydrogen generation and burn from potentially failing containment. Therefore, since this SAMA modification could eliminate hydrogen burns, the benefit was based on being equivalent to that associated with eliminating all hydrogen burns.

The benefit associated with reducing the hydrogen burn risk for each unit is defined by SAMA 96, "Provide post-accident containment inerting capability" (Reduced likelihood of hydrogen and carbon monoxide gas combustion). Therefore, the SAMA 96 benefit values were assigned as the SAMA 98 benefit for each unit (\$30.3K and \$25.8K at Unit 1 & 2, respectively). The cost of installing protective material (e.g., fire brick) beneath the reactor at Unit 1 was estimated as exceeding \$100K based upon the installation location (access and dose) and material costs. Performing analyses of the material addition to the containment for seismic and heat sink effects, as well as configuration control updates, were additional cost items.

The evaluation of SAMA 98 for Unit 2 used the Unit 1 minimum \$100K estimated cost since the plant configurations are nearly identical in the affected area beneath the reactors. Consistent with the Unit 1 evaluation, the \$100K cost was compared to the SAMA 96 benefit at Unit 2 (\$25.8K). The Expert Panel assigned "Excessive Cost" to the comparison for SAMA 98 as indicated in Table 6-1 of the ER, Attachment C-2, page C.2-90. However, this was done without actually identifying the estimated cost that was considered (\$100K). The question's reference to a lower cost for Unit 1 apparently results from the fact that no Unit 2 Phase II evaluation was performed for SAMA 98, and the use of Table 6-1 "Excessive Cost" implies higher cost values, although this was not the case.

#### RESPONSE SAMA-6.d

**d. For SAMAs 112 and 113 the implementation costs appear to be higher than expected. Provide the basis for the cost estimates for these SAMAs.**

As listed in the Unit 1 and Unit 2 Table 7-1 of the ER, Attachments C-1 and C-2, pages C.1-112 and C.2-102, SAMA 112 involves the addition of redundant and diverse limit switches to each containment isolation valve (CIV) to reduce the frequency of containment isolation failure and ISLOCAs (intersystem LOCAs). The individual benefits to each unit from eliminating all CIV failures are estimated to be \$5.8K at Unit 1 and \$20.1K at Unit 2. The Expert Panel estimated the cost of installation of redundant and diverse limit switches to each CIV would exceed \$1,000K per unit. The estimated cost at each unit was based in a conceptual design involving installation of additional safety-related valve limit switches, cables, conduit, supports and repairs to breached fire barriers, as well as associated drawing changes, design analyses, procedure changes, and training.

Also listed in Table 7-1 (for each unit – same ER pages), SAMA 113 involves the increase of leak testing of valves in the ISLOCA paths to reduce the frequency of ISLOCAs. The individual benefits to each unit from eliminating all ISLOCAs events are estimated to be \$9.9K at Unit 1 and \$135K at Unit 2. The Expert Panel estimated the cost of the increased leak testing would exceed \$1,000K per unit. The estimated cost at each unit was based on the additional testing being performed mid-cycle, since current improved standard technical specification (ISTS) surveillance requirements for reactor coolant system (RCS) pressure isolation valves (PIVs; see Specification 3.4.14) already require leak testing to be performed at least once each refueling outage. Therefore, increasing the frequency of testing would involve power replacement cost due to the mid-cycle shutdown at a rate of \$800K per day, in addition to the cost for labor and materials required to perform the testing.

#### RESPONSE SAMA-6.e

- e. The benefit of implementing a SAMA is estimated by summing the reductions in four major severe accident costs: offsite exposure cost, off-site economic cost, onsite exposure cost, and on-site economic cost. Table 7-1 provides the reduction in offsite dose and CDF but does not include the reduction in Offsite Economic Cost Risk (OECR). Provide the reduction in OECR for each SAMA.**

The evaluation results for benefits presented throughout the SAMA evaluation report in the ER, Attachment C, include Offsite Economic Risk as part of the total benefit. The development of the total benefit is explained in Attachment C-1 and C-2, Section 4, “Cost of Severe Accident Risk / Maximum Benefit.”

Table 7-1 of the ER, Attachments C-1 and C-2, starting on pages C.1-108 and C.2-99 contains the following information as noted in the RAI question:

- % Reduction in CDF
- % Reduction in Offsite Dose
- Benefit

Section 7.1.3 of NEI 05-01 defines the information needed for the Phase II analysis and defines “Benefit.” The evaluations are based on total benefit as described in the guidance and as provided in Table 7-1. Offsite Economic Cost Risk (OECR) is a component of the benefit for each SAMA, but is not required to be provided per NEI 05-01. However, Tables 6.E-1 and 6.E-2, shown below, provide the change in OECR for each SAMA evaluated in Phase II.



**Table 6.E-1**  
Unit 1 Change in Offsite Economic Cost Risk

<b>BV1 SAMA No.</b>	<b>Potential Improvement</b>	<b>Change in OECR</b>
1	Provide additional DC battery capacity	\$9.1K
2	Replace lead-acid batteries with fuel cells	\$9.1K
4	Improve DC bus load shedding	\$9.1K
5	Provide DC bus cross-ties	\$9.1K
6	Provide additional DC power to the 120/240V vital AC system	\$9.1K
13	Install an additional, buried off-site power source	\$51.9K
14	Install a gas turbine generator	\$261K
25	Install an independent active or passive high pressure injection system	\$15.2K
26	Provide an additional high pressure injection pump with independent diesel	\$15.2K
28	Add a diverse low pressure injection system	<\$1K
29	Provide capability for alternate injection via diesel-driven fire pump	<\$1K
37	Upgrade the chemical and volume control system to mitigate small LOCAs	\$31.3K
39	Replace two of the four electric safety injection pumps with diesel-powered pumps	\$15.2K
41	Create a reactor coolant depressurization system	\$31.3K
48	Cap downstream piping of normally closed component cooling water drain and vent valves	<\$1K
54	Increase charging pump lube oil capacity	<\$1K
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel	\$878K
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel	\$878K
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie	<\$1K
65	Install a digital feed water upgrade	\$21.8K
89	Improve SRV and MSIV pneumatic components	<\$1K
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	\$908K
96	Provide post-accident containment inerting capability	\$24.6K
98	Create a core melt source reduction system	\$24.6K
104	Improve leak detection procedures	\$5.7K
107	Install a redundant containment spray system	\$908K
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs	\$7.3K
112	Add redundant and diverse limit switches to each containment isolation valve	\$4.0K
113	Increase leak testing of valves in ISLOCA paths	\$7.3K

BV1 SAMA No.	Potential Improvement	Change in OECR
118	Improve operator training on ISLOCA coping	\$7.3K
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage	\$24.5K
122	Install a redundant spray system to depressurize the primary system during a steam generator tube rupture	\$24.5K
130	Add an independent boron injection system	\$4.3K
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS	\$4.3K
133	Install an ATWS sized filtered containment vent to remove decay heat	\$4.3K
136	Install motor generator set trip breakers in control room	\$4.3K
137	Provide capability to remove power from the bus powering the control rods	\$4.3K
147	Install digital large break LOCA protection system	\$5.7K
153	Install secondary side guard pipes up to the main steam isolation valves	<\$1K
155	Reactor Trip breaker failure—Enhance Procedures for removing power from the bus	\$4.3K
164	Modify emergency procedures to isolate a faulted ruptured SG due to a stuck open safety valve. This SAMA to provide procedural guidance to close the RCS loop stop valve to isolate the generator from the core and provide mechanical device to close a stuck open SG safety valve.	\$24.5K
165	Install an independent RCP Seal Injection system	\$878K
166	Provide additional emergency 125V DC battery capability	\$9.1K
167	Increase the seismic ruggedness of the emergency 125V DC battery block walls	\$910K
168	Install fire barriers for HVAC fans in the cable spreading room	\$92.5K
169	Improve operator performance. Operator starts Aux RW pump given offsite power is available.	\$2.1K
170	Improve operator performance. Operator starts portable fans & open doors in emergency switchgear room.	\$65.1K
171	Improve operator performance. Operator initiates Safety Injection.	\$2.1K
172	Improve operator performance. Operator initiates bleed and feed cooling given failure of prior actions to restore feedwater systems.	\$35.3K
173	Improve operator performance. Operator initiates makeup of RWST.	<\$1K
174	Improve operator performance. Operator trips RCPs during loss of CCR.	\$6.5K
175	Improve operator performance. Operator initiates depressurization of RCS given a general transient initiating event.	<\$1K
176	Improve operator performance. Operator initiates depressurization of RCS given a SGTR event.	<\$1K
177	Improve operator performance. Operator initiates cooldown and depressurization of RCS given a Small LOCA and failure of HHSI.	<\$1K
178	Improve operator performance. Operator aligns hot leg recirculation.	<\$1K
180	Reroute River Water pump power cable	\$19.7K
182	Reroute CCR pump or HHSI suction MOV cables	<\$1K
183	Reroute river water or auxiliary river water pump power and control cables	\$114K

BV1 SAMA No.	Potential Improvement	Change in OECR
184	Reroute river water or auxiliary river water pump power and control cables	\$32.7K
186	Add guidance to the SAMG to consider post-accident cross-tie of the two unit containments through the gaseous waste system	\$908K
187	Increase seismic ruggedness of the ERF Substation batteries. This item applies to the battery rack only and not the entire structure.	\$343K
188	Install a cross-tie between the Unit 1 and Unit 2 RWST	\$489K
189	Provide Diesel backed power for the fuel pool purification pumps and valves used for makeup to the RWST	\$489K
190	Reduce or eliminate the risk from control room fire CR1L1P. Provide fire barrier or mitigation inside connected control panels. (See Enclosure B for ER Unit 1 SAMA 190 information)	\$29.9K

**Table 6.E-2**  
Unit 2 Change in Offsite Economic Cost Risk

<b>BV2 SAMA No.</b>	<b>Potential Improvement</b>	<b>Change in OECR</b>
3	Add additional battery charger or portable, diesel-driven battery charger to existing DC system	\$1,010K
13	Install an additional, buried off-site power source	\$344K
14	Install a gas turbine generator	\$976K
17	Create a cross-tie for diesel fuel oil (multi-unit site)	\$23.6K
25	Install an independent active or passive high pressure injection system	\$13.4K
26	Provide an additional high pressure injection pump with independent diesel	\$13.4K
28	Add a diverse low pressure injection system	<\$1K
29	Provide capability for alternate injection via diesel-driven fire pump	<\$1K
37	Upgrade the chemical and volume control system to mitigate small LOCAs	\$53.4K
39	Replace two of the four electric safety injection pumps with diesel-powered pumps	\$13.4K
41	Create a reactor coolant depressurization system	\$53.4K
54	Increase charging pump lube oil capacity	<\$1K
55	Install an independent reactor coolant pump seal injection system, with dedicated diesel	\$886K
56	Install an independent reactor coolant pump seal injection system, without dedicated diesel	\$886K
64	Implement procedure and hardware modifications to allow manual alignment of the fire water system to the component cooling water system, or install a component cooling water header cross-tie	\$3.8K
65	Install a digital feed water upgrade	\$16.8K
78	Modify the startup feedwater pump so that it can be used as a backup to the emergency feedwater system, including during a station blackout scenario	\$1,181K
89	Improve SRV and MSIV pneumatic components	<\$1K
94	Install a filtered containment vent to remove decay heat. Option 1: Gravel Bed Filter; Option 2: Multiple Venturi Scrubber	\$1,779K
96	Provide post-accident containment inerting capability	\$20.4K
104	Improve leak detection procedures	\$4.4K
107	Install a redundant containment spray system	\$1,779K
111	Install additional pressure or leak monitoring instruments for detection of ISLOCAs	\$100K
112	Add redundant and diverse limit switches to each containment isolation valve	\$13.7K
113	Increase leak testing of valves in ISLOCA paths	\$100K
118	Improve operator training on ISLOCA coping	<\$1K
119	Institute a maintenance practice to perform a 100% inspection of steam generator tubes during each refueling outage	\$122K
130	Add an independent boron injection system	\$1.2K

BV2 SAMA No.	Potential Improvement	Change in OECR
131	Add a system of relief valves to prevent equipment damage from pressure spikes during an ATWS.	\$1.2K
133	Install an ATWS sized filtered containment vent to remove decay heat.	\$1.2K
136	Install motor generator set trip breakers in control room.	\$1.2K
137	Provide capability to remove power from the bus powering the control rods.	\$1.2K
153	Install secondary side guard pipes up to the main steam isolation valves.	\$1.0K
155	Reactor Trip breaker failure—Enhance Procedures for removing power from the bus	\$1.2K
164	Modify emergency procedures to isolate a faulted ruptured SG due to a stuck open safety valve. This SAMA to provide procedural guidance to close the RCS loop stop valve to isolate the generator from the core and provide mechanical device to close a stuck open SG safety valve.	\$64.8K
165	Install an independent RCP Seal Injection system	\$886K
169	Improve operator performance. Operator fails to align makeup to RWST - SGTR, secondary leak.	\$7.8K
170	Improve operator performance. Operator fails to manually trip reactor – ATWS.	<\$1K
171	Improve operator performance. Operator fails to realign main feedwater - no SI signal.	\$9.0K
172	Improve operator performance. Operator fails to initiate AFW following transient.	\$28.2K
173	Improve operator performance. Operator aligns spare battery charger 2-9 to 2-2.	\$3.4K
174	Improve operator performance. Operator aligns spare battery charger 2-7 to 2-1.	\$3.6K
175	Improve operator performance. Operator fails to initiate bleed and feed.	\$11.3K
176	Improve operator performance. Operator fails to trip RCP during loss of CCP.	\$4.0K
177	Improve operator performance. Operator fails to initiate bleed and feed.	\$1.0K
178	Improve operator performance. Operator fails to identify ruptured SG or initiate isolation.	\$13.9K
179	Reduce risk contribution from fires originating in Zone CB-3, causing a total loss of main feedwater and auxiliary feedwater with subsequent failure of feed and bleed	\$19.2K
180	Reduce risk contribution from fires originating in zone CT-1, causing a total loss of service water	\$132K
181	Reduce risk contribution from fires originating in zone SB-4, causing a total loss of normal AC power with subsequent failure of emergency AC power and station crosstie leading to station blackout	\$7.0K
183	Reduce risk contribution from fires originating in zone CV-3, causing failure of component cooling water (thermal barrier cooling) and service water with subsequent failure of reactor coolant pump seal injection	\$35.0K
184	Reduce risk contribution from fires in EDG building, fire initiator DG1L1A	\$107K
185	Reduce risk contribution from fires in EDG building, fire initiator DG2L1A	\$107K
186	Increase seismic ruggedness of the ERF Substation batteries. This item refers only to the battery racks, not the entire structure.	\$2.5K

<b>BV2 SAMA No.</b>	<b>Potential Improvement</b>	<b>Change in OECR</b>
187	Reduce risk contribution from internal flooding in cable vault area, CV-2 735', by reducing the frequency of the event or by improvements in mitigation of the resulting flooding.	<\$1K
188	Reduce risk contribution from internal flooding in Safeguards building, N&S (Source of flooding is a RWST line)	\$41.4K
190	Add guidance to the SAMG to consider post-accident cross-tie of the two unit containments through the gaseous waste system	\$1,779K

### **Question SAMA-7**

**There are a number of SAMA candidates for which implementation at a single unit would provide the intended benefits at both units, e.g., Unit 1 SAMAs 14, 186, 187, and 188, and Unit 2 SAMAs 14, 186, and 190. Identify all such “dual unit SAMAs” and describe the development of the implementation cost estimates for these SAMAs.**

### **RESPONSE SAMA-7**

Section 7.3 (for each unit) of the ER, Attachments C-1 and C-2, pages C.1-106 and C.2-97, discuss the shared (dual-unit) SAMAs. These sections identify the SAMAs by number for each unit and discuss the assessment of the SAMAs, considering the fact that they are shared by both units. Costs of these SAMAs were either confirmed or determined by the Expert Panel in a manner similar to the estimates made for all the other SAMAs (see RAI SAMA-6.a response). Section 7.3 for each unit discusses the evaluation of the pairs of related SAMAs, and the evaluations clearly indicate that the shared nature of the cost is properly considered in the evaluations.

It was noted during preparation of this response that listings for Section 7.3 were inadvertently omitted from the Table of Contents for both Attachment C-1 and Attachment C-2 of the ER.

### Question SAMA-8

The following information is needed to clarify that the potential for lower cost SAMA candidates were considered for the following SAMA candidates:

- a. Explain if lower cost alternatives were considered for Unit 1 SAMAs 183 and 184 (which both involve rerouting river water or auxiliary river water pump power and control cables), for example, partially re-routing one train of power cables and modifying the fire procedure(s) to manually control the river water or the auxiliary river water pump, or using rated fire blankets/barriers rather than rerouting cables. Verify that no lower cost alternatives are viable.
- b. Explain if lower cost alternatives were considered for Unit 2 SAMAs 179 and 180 (which involve control room and cable tunnel fires, respectively), for example, partially re-routing or protecting one train of service water combined with procedures to allow manual local actions. Verify that no lower cost alternatives are viable.
- c. SAMA 54 (Unit 1), and SAMAs 55, 56, and 165 (Units 1 and 2) are focused on reducing the likelihood of RCP seal LOCAs. However, there could be other lower cost SAMAs such as adding a dedicated self-contained diesel driven pump for seal cooling or cross-connecting the chemical and volume control system (CVCS) from the opposite unit for RCP seal injection. Verify that no lower cost alternatives are viable.

### RESPONSE SAMA-8.a

- a. Explain if lower cost alternatives were considered for Unit 1 SAMAs 183 and 184 (which both involve rerouting river water or auxiliary river water pump power and control cables), for example, partially re-routing one train of power cables and modifying the fire procedure(s) to manually control the river water or the auxiliary river water pump, or using rated fire blankets/barriers rather than rerouting cables. Verify that no lower cost alternatives are viable.

During the development of the response to this question, it was identified that Unit 1 Table 7-1 and Table 8-1 of ER, Attachment C-1, pages C.1-116 and C.1-125, respectively, contained a typographical error in the "Cost" column for SAMA 184. The Cost column entries read ">\$2,000", but the entries in both tables should read ">\$2,000K". See Enclosure B for ER revisions to Unit 1 SAMA 184 information.

In response to the question, the potential for lower cost alternatives to each of the SAMAs was considered in the Expert Panel discussions as part of the evaluation process.



The Expert Panel consisted of staff members that are experts in system design, plant operation, maintenance, modifications, and instrumentation and control. The guidance provided to the Expert Panel during the initial meeting was that they consider the SAMA statement and that they consider the purpose of the SAMA, including in their consideration alternatives that would accomplish the purpose of the SAMA.

The following is an excerpt from the Expert Panel Agenda. Note Item V.g directing the discussion toward alternative mitigative strategies or changes.

- “V. Discussion of Refined SAMA Case Analysis (Cost Evaluations) – Expert Panel
  - a. Describe the Change Modeled
  - b. Reiterate the Purpose of the Change
  - c. EP Discuss the Suggested Change
  - d. EP Discuss the Cost of the Change
  - e. EP Concur on Cost Estimate for the Change
  - f. If the Cost Estimate is within ~5% of Benefit Defer to Detailed Cost Estimation by BV.
  - g. EP Discuss other alternative mitigation strategies or changes.”

The Expert Panel process required solicitation and discussion of lower cost alternatives from Panel members. The cost estimates developed by the Expert Panel, therefore, considered the viability of the lower cost options, while also recognizing that each specific design alternative suggested would have benefits that are lower than the bounding benefit values provided for each SAMA analysis.

During the Expert Panel review of SAMAs 183 and 184, no lower cost alternatives were identified to fully address the suggested alternatives.

The two suggested alternatives provided in this RAI are viable solutions for at least partially reducing risk, but neither are cost beneficial. The first alternative suggestion of rerouting a single train of power cables and modifying procedures to manually control pumps would reduce the risk associated with the CS-1 (cable spreading room/cable tray mezzanine) and NS-1 (normal switchgear room) fire scenarios. However, rerouting even one train of cabling is still very costly, and well above the stated benefits listed in the ER, Attachment C-1 for Unit 1, Tables 7-1 and 8-1, of \$163K and \$50K, respectively. The engineering change package (ECP) development costs (including design conceptualization efforts, design drawing changes, ECP internal reviews, fire PRA/hazards analyses, seismic loading re-analyses, licensing document updates, etc.) alone would exceed the stated benefits. Installation costs for rerouting the associated safety-related power cables would potentially include all or most of the following: fire barrier breaches and repairs, installation of new splice boxes, installation of new conduit and/or trays, and installation of additional lengths of power cabling. Total costs would

approach those stated in Tables 7-1 and 8-1 of >\$2,000K (see Unit 1 Table 7-1 and 8-1 correction discussion at the beginning of this section). The second alternative suggested of using rated fire blankets/barriers would incur ECP development costs similar to those described for the first suggested alternative, and exceed the stated benefits; installation costs would, again, only further add to the non-cost beneficial result.

Concerning modification of fire procedures to cover manual control of pumps, existing fire emergency procedures (FEPs) have already been enhanced to provide actions to locally start a river water pump. For SAMA 183, a fire in area CS-1 requires evacuation of the Control Room. The FEPs for this scenario now include steps to locally control the Train B river water pump. For SAMA 184, the FEP for a fire in area NS-1 contains a step to locally verify that the breaker to start Train B river water pump is closed. As an alternative, the NS-1 related FEP also contains a step to locally start the swing (C) river water pump on Train B.

#### RESPONSE SAMA-8.b

**b. Explain if lower cost alternatives were considered for Unit 2 SAMAs 179 and 180 (which involve control room and cable tunnel fires, respectively), for example, partially re-routing or protecting one train of service water combined with procedures to allow manual local actions. Verify that no lower cost alternatives are viable.**

Lower cost, partial alternatives were considered in the evaluation by the Expert Panel (EP) as discussed above in RAI Question SAMA-8.a, and none were found to be both viable and cost beneficial. The two suggested alternatives in this RAI Question were considered viable, but not cost beneficial.

Unit 2 SAMA 179 specifically deals with reducing risk contribution from fires originating in the Zone CB-3 (i.e., the main control room [MCR] area) causing a total loss of main feedwater and auxiliary feedwater with subsequent failure of [RCS] feed and bleed. Rerouting of cabling (or even adding fire blankets/barriers) to prevent a fire in the MCR area from affecting all feedwater train controls, which are located in common MCR control board areas, was considered by the EP to be impractical and well beyond the benefit of \$34.4K listed in Unit 2 Tables 7-1 and 8-1 of the ER, Attachment C-2, pages C.2-105 and C.2-115, respectively. The primary plant design feature already in existence that ensures safe plant shutdown in the unlikely event of a MCR fire scenario is the transferability of limited controls to the Alternate Shutdown Panel, which is located outside of the MCR area in the Unit 2 Cable Vault Area. No further cost-effective improvements to this design feature were conceptualized by the EP.

Unit 2 SAMA 180 specifically deals with reducing risk contribution from fires originating in the Zone CT-1 (i.e., the Cable Tunnel Area which physically connects the Control

Building underground with the Unit 2 Cable Vault Area) causing a total loss of service water. Essentially all Unit 2 safety- and nonsafety-related control and instrumentation cabling with connections between MCR controls and associated plant equipment passes through Zone CT-1. One train (i.e., Train B, also known as the "purple train") is already hardened with fire wrapping. Rerouting control cabling or hardening the other train (i.e., Train A, also known as the "orange train") for service water only is impractical and well beyond the benefit of \$202K listed in Unit 2 Tables 7-1 and 8-1 of ER, Attachment C-2, pages C.2-105 and C.2-115, respectively. The engineering change package (ECP) development costs (including design conceptualization efforts, design drawing changes, ECP internal reviews, fire PRA/hazards analyses, seismic loading re-analyses, licensing document updates, etc.) alone would exceed the stated benefit. Installation costs would only further add to the non-cost beneficial result.

#### RESPONSE SAMA-8.c

- c. SAMA 54 (Unit 1), and SAMAs 55, 56, and 165 (Units 1 and 2) are focused on reducing the likelihood of RCP seal LOCAs. However, there could be other lower cost SAMAs such as adding a dedicated self-contained diesel driven pump for seal cooling or cross-connecting the chemical and volume control system (CVCS) from the opposite unit for RCP seal injection. Verify that no lower cost alternatives are viable.**

Addressing Unit 1 SAMA 54 separately, this SAMA specifically deals with increasing charging pump lube oil capacity in order to increase the time before pump failure due to lube oil overheating in the loss of cooling water sequences. Due to the very low benefit of <\$1K listed in Unit 1 Tables 7-1 and 8-1 of ER, Attachment C-1, pages C.1-110 and C.1-122, respectively, no feasible lower cost alternative was identified by the EP. The benefit is well below the assumed minimum cost associated with development and implementation of an integrated hardware modification package of \$100K, as described in Section 7.2 of ER, Attachment C.

Concerning the broader issue of considering lower cost alternatives, as discussed above in the response to SAMA-8.a, lower cost alternatives were considered in the SAMA evaluations performed by the Expert Panel (EP), and none were found to be both viable and beneficial for the remaining SAMAs listed above.

The primary design conceptualized to address SAMAs 55, 56 and 165 at both units was the installation of independent, diverse and redundant seal injection systems at each unit. These independent systems would be seismically qualified, draw their make-up water from the associated Unit's refueling water storage tank (RWST), and have dedicated seal injection pumps with independent power supplies and independent connections to the RCP seal injection lines. This conceptual design formed the basis for the cost of >\$4,000K listed in Unit 1 & 2 Tables 7-1 and 8-1 for all three SAMAs, and

which significantly exceeded the associated benefits for eliminating all RCP seal LOCA events of \$1,303K and \$1,358K for Unit 1 & 2, respectively.

Other lower-cost alternatives considered included the installation of a skid mounted diesel generator as a backup power source to the charging pumps, and modifications to the safety injection system hydro test pump at each unit. However, it was considered that these changes would be only partially beneficial in that they would still rely on the functionality on the charging pumps and/or their normal seal injection flow paths, and not satisfy the overall goal of providing an independent seal injection capability that would eliminate all RCP seal LOCAs. Also, the stated benefits of \$1,303K and \$1,358K would not necessarily fully apply in this case. Therefore, these alternatives were not pursued past the Phase II analysis process.

Addressing the suggested alternatives in the question, the addition of dedicated, self-contained diesel-driven pumps at each unit would provide a partial solution that would necessitate outdoor installations with remote manual or automatic starting capability, long runs of new seismically-supported piping, and new connections to each RWST. Also, valves that connect with the existing RCP seal injection paths would require fast-acting operators with remote operation capability, since backup seal injection flow must occur quickly when called upon to prevent any significant interruption in cooling water flow and undesired thermal shocks to the seals. In the opinion of the EP, the associated costs would significantly exceed any partial benefit.

Cross-tying the units' CVCS systems would provide another partial solution to the loss of RCP seal cooling concern. The cost of the cross-tie is not low, since the units occupy physically separate locations; the installation would require significant excavation effort to run piping from one unit to the other. The BVPS plants are substantially different than other two-unit stations that can be cross-tied by simply running interior piping from one building to the contiguous structure of the other unit. Excavating and burying piping, with the potential interference with existing buried piping/ducts, is expensive. The cost associated with developing a cross-tie between the two units is estimated by the EP to also significantly exceed any partial benefit.