APS

A subsidiary of Pinnacle West Capital Corporation

Palo Verde Nuclear
Generating Station

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November 10, 2009

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

Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Docket Nos. STN 50-528, 50-529 and 50-530 Response to September 30, 2009, Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the Review of the PVNGS License Renewal Application, and License Renewal Application Amendment No. 2

By letter no. 102-05937, dated December 11, 2008, as supplemented by letter no. 102-05989, dated April 14, 2009, Arizona Public Service Company (APS) submitted a license renewal application (LRA) for PVNGS Units 1, 2, and 3. By letter dated September 30, 2009, the NRC issued a request for additional Information (RAI) related to the PVNGS LRA. Enclosure 1 contains APS's response to the September 30, 2009, RAI. Enclosure 2 contains LRA Amendment No. 2 changes to reflect the RAI responses.

Commitments in this submittal are described in the LRA Amendment 2 changes in Enclosure 2. Should you need further information regarding this submittal, please contact Russell A. Stroud, Licensing Section Leader, at (623) 393-5111.

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

Executed on 11/10/09
(date)

Sincerely,

[signature]

DCM/RAS/GAM/

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NRR

Enclosures:

1. Response to September 30, 2009, Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the Review of the Palo Verde Nuclear Generating Station License Renewal Application

2. Palo Verde Nuclear Generating Station License Renewal Application Amendment No. 2

cc:	E. E. Collins Jr.	NRC Region IV Regional Administrator
	J. R. Hall	NRC NRR Project Manager
	R. I. Treadway	NRC Senior Resident Inspector for PVNGS
	L. M. Regner	NRC License Renewal Project Manager

ENCLOSURE 1

Response to September 30, 2009, Request for Additional Information Regarding Severe Accident Mitigation Alternatives for the Review of the Palo Verde Nuclear Generating Station License Renewal Application

NRC RAI 1.a

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Section D.2.3 provides a detailed description of the PRA model changes made since the IPE (individual plant examination) Level 1 model. For each "major" version of the PRA model since the IPE, identify the model changes listed in Section D.2.3 that correspond to each version, identify the model changes that most impacted the change in core damage frequency (CDF) and large early release frequency (LERF), and provide the CDF. Two of the "major" PRA versions should be the version that was peer reviewed in November 1999 and Revision 15 used in the SAMA evaluation.

APS Response to RAI 1.a

The PVNGS PRA Model revision 15 was used to prepare the SAMA analysis. The PRA model peer reviewed in November 1999 by the Combustion Engineering Owners Group (CEOG) was Revision 3. The significant model changes (since Revision 0) are listed below. Up to Revision 14, the PVNGS PRA model contained a partial Level 2 model that only calculated LERF. That partial Level 2 model was not sufficient to prepare the SAMA analysis. PVNGS upgraded the Level 2 model for use in the SAMA analysis using Westinghouse guideline WCAP-16341-P (Ref. WEST 2005 in ER Section D.11).

Revision 0 of the PVNGS PRA model documented Level I model changes made to enhance the capability to model specific plant maintenance configurations, such as for use in generating the Plant Configuration Risk Indicator Matrix (PCRIM) and for evaluating transition risk (shutdown, cool-down, heat-up and Mode 4 steady-state). Also included in the model revision was correction of modeling errors, one of which resulted in greater than a 5% change (decrease) in CDF, specifically that station blackout sequences did not properly credit the short-term availability of the steam-driven auxiliary feedwater pump. The CDF for this revision was 3.93E-05/yr.

Revision 1 updated the mean values of the basic events used in the importance analysis and corrected an error in an initiating event frequency. This update of the basic event mean values did not impact solution of the model, with the exception of importance analysis. The loss of condensate pump initiator frequency increased relative the previous model, as a result of failing to update a table in the recovery database following a single change made to the model documented in Revision 0. This change resulted in no detectable increase in the overall CDF.

Revision 2 of the PVNGS PRA model was developed to correct the application of a human reliability analysis (HRA) for stationing an operator to take manual control of the downcomer feedwater isolation valves on loss of nitrogen. The positioning of this HRA in the logic caused it to be inappropriately applied to the downcomer feedwater regulating valves. Several other changes were made as enhancements and to correct less significant errors. The CDF for this revision was 4.18E-05/yr.

Revision 3 of the PVNGS PRA model added the downcomer block valve to the model, since it is required to stop flow through a failed "as-is" downcomer feedwater control valve when using the downcomer bypass valve. The station blackout initiating event frequency was also updated with new data. In addition, Revision 3 incorporated a change in test intervals associated with engineered safety features actuation system (ESFAS) surveillance tests. The frequency of these ESFAS relay tests was changed from two months to nine months (staggered) at the time the Improved Standard Technical Specifications were adopted in 1998. Additional changes included recalculation of certain control circuit and common-cause values, reflecting increases in test intervals for many auxiliary feedwater and safety injection valves, recalculation of the engineered safety features (ESF) switchgear room heating, ventilation and air conditioning (HVAC) room cooler failure probability, changes to loss of coolant accident (LOCA) frequencies, and correction of restore-after-maintenance (RM) events that were not treated consistently. The CDF for this revision was 5.79E-05/yr.

Revision 4 of the PVNGS PRA model included the incorporation of extensive changes to the model to allow equipment to be taken out of service while still maintaining correct cutsets and recoveries. The model changes also included the impact of a system out of service (OOS) whether the system is modeled as a complex initiator or a mitigation fault tree. These changes were made to support updating the PCRIM in response to 10 CFR 50.65(a)(4). Revision 4 also included removing credit for 125 VDC power when the bus battery is failed since the charger alone is incapable of providing the peak load immediately after a plant transient. Revision 4 included an update of the initiating event data. Overall, the changes made to the model resulted in a CDF increase of 20% to 6.96E-5/yr. LERF increased 35% to 5.82E-6/yr.

Revision 5 of the PVNGS PRA model was associated with documentation, not model changes. There was no change in CDF in this revision.

Revision 6 of the PVNGS PRA model updated the revision numbers of various references used by the PRA model. There was no change in CDF in this revision

Revision 7 of the PVNGS PRA model accomplished several changes:

- Added recovery rules so that recoveries can be applied within the solution process versus manually applying them after the model solution.
- Incorporated the fire PRA model.
- Corrected modeling of Class 1E battery common-cause.
- Credited the check valve for penetration 41 (charging system flow path).
- Corrected modeling of containment sumps; failure to isolate the refueling water tank (RWT) on either train fails both trains of recirculation.
- Corrected modeling of control circuit after receipt of a safety injection actuation signal (SIAS) for the non-class motor-driven auxiliary feedwater pump.
- Added modeling of the back-up power supplies for the new digital feedwater control system.

- Added new LERF event trees to properly account for boundary conditions that are not transferred from the CDF results.

The overall effect on the internal events results is a CDF decrease from 6.96E-5/yr to 2.13E-5/yr and LERF decrease from 5.82E-6/yr to 2.57E-6/yr. The calculated CDF from fire events alone is 6.98E-6/yr and LERF is 3.37E-7/yr. This gives a combined CDF due to internal events and fire of 2.77E-5/yr, 23% of which is due to fire initiators. The combined LERF is 2.83E-6/yr, 9% of which is due to fire initiators.

Revision 8 was a documentation update and had no impact on the CDF.

Revision 9 was a major update that incorporated 237 open impacts identified by the PRA maintenance program. Documentation of the issues and the resolutions are addressed in the PRA Impact Database.

Major changes made by this update included crediting alternate paths to carry power to the mitigating systems, addressing internal comments and CEOG peer review comments. The CDF for this revision was 1.75E-05/yr.

Revision 10 incorporated 115 model impacts identified by the PRA maintenance program. Documentation of the issues and the resolutions are addressed in the PRA Impact Database.

Major changes made by this update included correction of loss of off-site power (LOOP) non-recovery probabilities, incorporation of new LOCA success criteria, maintenance activity logic changes, and unavailability data updating.

The internal events CDF increased from 1.75E-5/yr to 1.77E-5/yr. The internal events LERF also increased from 2.38E-6/yr to 2.42E-6/yr.

The fire CDF decreased from 5.48E-6/yr to 5.06E-6/yr. The fire LERF decreased from 2.52E-7/yr to 2.33E-7/yr.

The combined CDF decreased slightly from 2.27E-5/yr to 2.25E-5/yr. The contribution of fire to the total changed from 24% to 21%. The combined LERF increased slightly from 2.60E-6/yr to 2.63E-6/yr. The contribution of fire to the total changed from 7% to 7.1%.

Revision 11 incorporated changes to LOOP non-recovery probabilities, eliminated double LOOP recoveries and removed some unused basic event IDs. The combined CDF for this revision was 1.75E-05/yr.

Revision 12 incorporated changes to fire event trees, corrected basic event names that did not meet 16-character naming convention, and incorporated changes for new Rudd Transmission Line installation. The CDF for this revision was 1.43E-05/yr.

Revision 13 implemented the following changes:

- Reactor coolant pump (RCP) seal modeling changes to be consistent with the NRC RCP seal leakage model.
- Removal of pump room HVAC dependencies based on room heatup calculations.
- Correcting the power supply logic for electric driven fire pump.
- Applying consistent application of rules for deleting invalid cutsets.
- Incorporation of not-logic for using in the EOOS (equipment out of service) risk monitor.
- Deleting credit for atmospheric dump valves in a steam generator tube rupture event.
- Updating the test intervals for auxiliary feedwater valves and relays.
- Updating turbine bypass modeling logic.
- Correcting instrument air power supply logic.
- Addressing potential for an anticipated transient without scram in a steam generator tube rupture and small loss of coolant accident event.
- Correcting hot leg injection logic.
- Updating the human reliability probability for containment spray header flange fail-to-restore.
- Correcting blowdown path modeling.
- Crediting use of the reactor makeup water tank as a water source for the auxiliary feedwater.
- Including induced steam generator tube rupture potential following a steam line break.
- Addition of turbine cooling water isolation valves for IA compressor coolers.
- Updating values for feedwater isolation valve recovery actions.
- Providing justification for not modeling HVAC in charging pump rooms.

Revision 13 also removed RCP seal leakage/rupture events, credited shortened test intervals for certain auxiliary feedwater (AFW) system valves and relays, corrected hot leg injection modeling, corrected SG blowdown path modeling, and updated steam line break modeling to include pressure-induced steam generator tube rupture.

In addition, several changes were made to the fire model. Power supply logic for the electric fire pump was corrected. Several issues identified in the CEOG peer review were addressed including adjusting dependence of the steam bypass control and reactor power cutback systems on non-vital AC and station DC.

The net impact to CDF and LERF was as follows: internal events CDF decreased 11% from 1.43E-5/yr to 1.27E-5/yr; internal events LERF decreased 32% from 2.32E-6/yr to 1.57E-6/yr; fire CDF decreased 3% from 4.26E-6/yr to 4.15E-6/yr; fire LERF decreased 5% from 2.00E-7/yr to 1.90E-7/yr; combined CDF decreased 9% from 1.81E-5/yr to 1.64E-5/yr; combined LERF decreased 30% from 2.49E-6/yr to 1.74E-6/yr.

Revision 14 incorporated a total of 103 model impacts to the internal events and fire models. This resulted in two major global changes:

- Failure data update.
- Common-cause methodology update converting to the "alpha" model vs. a combination of the multiple Greek letter and binomial failure models, as well as utilization of Risk Spectrum's software common-cause modeling capability.

Introduction of the above impacts and quantification were broken down into three phases: 1) non-data/non-common-cause changes, 2) data changes, and 3) common-cause modeling changes. Results of the overall changes were:

- The internal events CDF increased to 1.39E-5/yr.
- The fire CDF was minimally impacted by the changes.

All Category B CEOG peer review comments concerning data and common-cause were closed by this revision. There were no remaining open Category A or B comments after this revision.

Revision 15 made the following significant modeling changes:

- Elimination of diesel-generator (DG) and pump control circuit faults; DG and pump failures now include command faults.
- Credit feeding either steam generator (SG) after steam generator tube rupture (SGTR) and removing alternate feedwater (AltFW) to simplify the modeling; this resulted in a significant LERF reduction.
- Reconstruction of SGTR top logic for cooldown and depressurization, including recalculation of some human reliability analyses (HRAs) and elimination of several others.
- Made occurrence of main steam isolation signal (MSIS) with SGTR conditional upon operator failure to control SG level rather than assuming it always occurs.
- Adoption of a more reasonable tank failure rate for the condensate storage tank (CST) and refueling water tank (RWT), along with removing the reactor water makeup tank (RWMT) as a back-up to the CST.
- Adoption of CE nuclear steam supply system (NSSS)-specific instrument failure probabilities, which resulted in elimination of common-cause modeling of RWT level sensor/transmitters, RCS pressure sensor/transmitters and bistables.
- Deletion of HVAC dependence for pumps.
- Removed HRA for overriding MSIS in order to use non-safety FW pump.
- Removed HRA for overriding MSIS and opening an MSIV to steam to the condenser. It is not possible to open an MSIV without first equalizing pressure across it. It was determined not worthwhile to add modeling for the MSIV bypass valves, which would actually be used for steaming instead of the MSIV.
- Adjusted auxiliary feedwater (AFW) and alternate feedwater (AltFW) HRAs for availability of main feedwater (MFW) for up to seven hours post-trip. This was done to more accurately represent plant operation, as well as attempt to provide

more margin for the electric AFW pumps in the mitigating systems performance index.

CDF decreased from 1.39E-5/yr to 5.07E-6/yr due primarily to elimination of the HVAC dependence of the ESF pumps, and crediting main feedwater for up to seven hours post-trip. Smaller contributors were the lower tank failure rate and use of lower failure probabilities for instrumentation.

The new PVNGS Level 2 model (December, 2007) was used in the SAMA analysis, and is described in Section D.2 of the ER.

Revision 16 was issued after the SAMA analysis was submitted to the NRC. The APS response to RAI 1.b below describes the potential Revision 16 impact on SAMA. Revision 16 made the following significant modeling changes:

- Restructured the station blackout event tree incorporating power recovery into the steam generator heat removal loss (SGHR_L) function event.
- Changed the DG failure modes from start and run failures to (1) start and come up to voltage and speed failures, (2) close in failures, (3) load and run for one hour failures, and (3) run for greater than one hour failures, in order to align with mitigating systems performance index (MSPI) program.
- Updated recovery rules associated with AFW vs. AltFW and those with power recovery for the new DG failure modes.
- Credited MFW for the full 24-hour mission time, as well as recovering loss of all feedwater with restarting MFW. The MFW pumps are turbine-driven pumps and had not been analyzed for long term operation at low flow. An engineering analysis was performed to support crediting MFW for up to 24 hours.
- Updated MSPI system unavailability parameters.

NRC RAI 1.b

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Section D.2.1.4 states that the Palo Verde Nuclear Generating Station (PVNGS) PRA model Revision 15 used for the SAMA analysis reflects PVNGS as designed and operated up to August 2008. Identify any changes to the plant (physical and procedural modifications) or open PRA issues identified since August 2008 that could have a significant impact on the results of the PRA and/or the SAMA analyses. Provide a qualitative assessment of their impact on the PRA and on the results of the SAMA evaluation.

APS Response to RAI 1.b

The PVNGS Level I PRA model (internal events and fire model) Revision 16 was completed after August 2008. A description of the major changes is provided in the APS response to RAI 1.a above. The APS PRA staff periodically reviews plant procedure changes and plant design modifications to assess their impact on the PRA level I and LERF models. There were no numerical changes to the model in Revision 16 from procedure changes or design modifications. The importance results from Revision 16 were compared with those from Revision 15 (used for SAMA analysis in Table D.5-1). Use of Revision 16 of the PRA would not have resulted in any new entries in Table D.5-1. The CDF in the Revision 16 PRA model decreased largely due to the additional credit taken for longer operation of main feedwater following a transient (as supported by engineering analysis). Use of Revision 16 would have the effect of lowering the overall man-rem consequences from those calculated using revision 15. A review of plant modifications since Revision 16 has not identified any potentially risk significant changes in CDF or LERF.

NRC RAI 1.c

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Section D.2.1.9 states that a Combustion Engineering Owner's Group (CEOG) peer review was performed on the PVNGS PRA model in November 1999. Identify the version of the PRA model that was reviewed in this peer review and describe the scope of the peer review including whether the Level 2 and 3 model was included in the review. Provide a brief description of the results of this peer review and the status of the findings. Provide an assessment of the impact of any unresolved findings on the results of the SAMA analysis.

APS Response to RAI 1.c

The CEOG peer review process was conducted in 1999 on PVNGS PRA model Revision 3. It employed a team of PRA and system analysts, each with significant expertise in PRA development and PRA applications. The team utilized a set of checklists covering eleven key technical areas as a framework within which to evaluate the scope, comprehensiveness, completeness, and fidelity of the PRA model being reviewed. These eleven key areas are:

1. Selection of initiating events.
2. Accident sequence analysis.
3. Thermal hydraulic analyses.
4. Systems analyses.
5. Data analysis.
6. Dependency analysis.
7. Human reliability analysis.

8. Structural analysis.
9. Quantification process.
10. LERF analysis (not a complete Level 2 or 3 model).
11. PRA maintenance and update process.

The peer review team found PRA maintenance and update, LERF analysis, structural analysis of containment fragility, and the thermal hydraulics analyses to be strengths. The following is a summary of the significant peer review comments:

- *Perform independent review of all assumptions.*
  This observation was closed by performing independent reviews of system assumptions on a system by system basis. This approach allowed PRA engineers to select for their reviews the more familiar systems.

- *Demand failures for station batteries should be considered for modeling.*
  This finding was incorporated. Also, the service life time for batteries was adjusted down based on engineering analysis.

- *Throttling of Auxiliary Feed water valves not modeled except for initial opening.*
  This finding was incorporated by expanding the governing operator manual actions (HRA) to include additional steps of throttling.

- *Failure mode of turbine-driven AFW pump due to condensate in the steam line not modeled.*
  This observation was evaluated with consideration to the associated plant modification. There were no AFA-P01 pump failures due to condensate since January 1, 1995.

- *The common cause failure factors for EDG failure to start and AFW pumps failure to run are lower than those used in the INEEL report.*
  This finding was closed by re-working all the CCF modeling in the PRA in accordance with NRC data, alpha factors, and method presented in NUREG/CR-5485.

- *Loss of offsite power (LOOP) frequency and duration is not current.*
  This finding was closed by (periodically) updating the LOOP study in accordance with the biennial EPRI report on industry LOOP data. Also, the LOOP frequency was subdivided into segments associated with plant-centered LOOP, Switchyard-centered LOOP, Grid-related LOOP, and Weather-related LOOP.

- *The HRAs modeled in the PRA did not describe operator involvement and did not show the modeling of pre-initiators associated with miscalibration of some critical sensors.*
  This finding was incorporated by adding sections on operator reviews and insights, adding several miscalibration sections, and by using the latest revision of the guidelines and software in the HRA calculator.

- *The PRA model documentation did not show sufficient analysis of HRA dependencies in the minimal cutsets.*
  This finding was closed by updating the method used for HRA dependencies and the application of recovery HRAs. Also, cross system linkages were added in the model to link HRAs with components.

- *The application of local recovery of AFA-P01 failure to start did not separate the estimated fractional value of non-recoverable failures.*
  This finding was evaluated by a model impact and the results were incorporated with the estimated fraction of non-recoverable failures.

- *There were insufficient uncertainty analyses performed.*
  This observation was evaluated for feasibility of performing an overall uncertainty analysis on CDF and LERF. The model (with over 3200 basic events) is much too large for such a meaningful analysis. Each model application identifies the parameters relevant to uncertainties and includes the resulting impacts of these uncertainties. The PRA data used in the model are mean values with bounding limits at the 5^{th} and 95^{th} percentiles.

- *The PRA documentation did not provide a guidance to re-evaluate prior PRA applications when significant model changes are implemented.*
  This observation was closed by establishing a clear procedural guideline for re-evaluating prior PRA applications when the model is significantly changed. Several application documents (such as MOV and AOV risk ranking) are routinely revised after each significant model update.

- *Loss of multiple independent vital 125 v DC and 120 v AC busses are not considered as initiators.*
  This observation was analyzed and closed out on the basis that each of the four individual buses is treated as initiator. And, all DC and AC buses are modeled for their common cause failure in accordance with the NRC guidelines and alpha factors in NUREG/CR 5485.

- *The PVNGS PRA lacks an internal flooding model.*
  The category of this observation was elevated at PVNGS (from C to B). It was compared with other needed model expansions. PVNGS is a post 1975 plant. Its design and construction incorporated most known elements of internal flood impacts. Currently, an internal flood model is at the beginning stages of its development.

In addition to the comments listed above, the PVNGS PRA model benefited from the CEOG peer review by adding numerous comments related to size and treatment of small LOCA, reactor vessel rupture event, and cooldown in MAAP4 analysis based on atmospheric dump valve openings (not steam bypass system).

All the CEOG Category A and B peer review comments have been addressed.

NRC RAI 1.d

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Identify and discuss any additional internal and external reviews of the Level 1 and Level 2 PRA models. Describe any significant review comments, their resolution, and the potential impact of any unresolved comments on the results of the SAMA analysis.

APS Response to RAI 1.d

A. ERIN Engineering review of Revision 6 PRA model in February/March, 2001

The objectives of this evaluation were to:

- Evaluate the technical issues identified in the PSA certification peer review and provide an independent evaluation of the importance of the issues in comparison with those identified in other industry PSA peer review programs. Identify new issues and propose changes to the issue priorities as appropriate.
- Evaluate the actions taken and responses to the peer review issues by the PRA team and provide an assessment of the extent to which the issues have been resolved in the current PRA update.
- Evaluate the PVNGS PRA update procedures and process and recommend enhancements that could be useful for current and future applications of the PVNGS PRA program.

There are no open items from this ERIN review. The over-all ERIN assessment result was summarized by the assessors as follows:

> Although the purpose of the peer review process is not to assign a single grade to the PRA as a whole, the ERIN review team presents some high level comments and observations in this section. It is important to note that the opinions and observations presented here are based on a review of eleven peer review technical elements (quantification, initiating events and PRA maintenance and update) and on the level A and B facts and observations identified by the CEOG peer review team.
>
> Based on the limited scope review, ERIN placed the PVNGS PRA in the top 25% of all PRAs in the US. The documentation of the PRA model configuration control and the linking of the documentation to PRA model elements were the best that they had seen for these attributes and would likely represent an industry

best practice. ERIN indicated the quality of the PRA supports the current Palo Verde PRA applications including SAMA analysis. Overall the PRA quality of each element was assessed capable of supporting risk significance evaluations with deterministic input (i.e., Grade 3) contingent on the resolution of the Category A and B observations identified in the CEOG peer review as augmented in their independent review. ERIN was aware of no plant that has been subjected to this peer review process which has not had identified some Category A and B issues that would need to be addressed in risk informed applications.

B. RELCON-AB Evaluation of revision 7 PRA model in August, 2001

This RELCON-AB evaluation covered the following model attributes:

- Use of attributes, Common Cause Failure modeling, Diamond Basic Events, Gates and FT pages, Coding system, Fault Trees, Boundary conditions sets, Function Events, Post Processing of cutsets, Event Trees, Fire modeling, Truncations, Exchange Events, Template Events, Reliability models, Success Criteria, and Out Of Service modeling.

There are no open items from this RELCON review.

C. ERIN Engineering Fire PRA Review associated with Revision 10 of the PRA model in February, 2003

A peer review of the Fire PRA for PVNGS was performed by ERIN Engineering in February 2003. The objectives of this peer review were to assess:

1. Transparency of documentation in supporting a clear understanding, including key assumptions;
2. Use of acceptable methodology in comparison with current state-of-the-art and industry practices;
3. Application is free of obvious errors or misapplications; and
4. Identify deficiencies and provide recommendations for future enhancements.

Overall, the review determined that the Fire PRA was well documented, complete, and comprehensive. A few specific issues were noted and are discussed further in the report. The resolution of these issues could cause the CDF to increase, while others would tend to cause the CDF to decrease. There are no open items from this review.

The key conclusions of the review were:

1. The analysis documentation is comprehensive, detailed, and complex. The complexity is reflective of the analysis rigor rather than a weakness. The preparation of a Topical White Paper that serves as a 'roadmap' for the user could be a great benefit for a 'new' user.

2. The crediting of automatic fire suppression actuation appears to have been assigned a much greater benefit than is warranted. This benefit is reflected in the assignment of values for the TRIP node. Depending on the specific fire compartment under consideration, a higher value or perhaps a 1.0 may be more appropriate.

3. The analysis relies greatly on many of the postulated plant fires not causing a plant trip or otherwise creating a demand for prompt shutdown. Since Fire PRAs typically do not explicitly identify circuits associated with PCS, EHC, or RPS, the ability to determine with certainty that a plant trip does not occur is difficult. The peer review results indicate that some scenarios should have considered a higher potential for fire-induced plant trip/shutdown.

There are no Category A or B comments open from this ERIN review.

D. Development and Reviews of the PVNGS Level 2 Model Used for SAMA Analysis

The guidelines used for development of the Level 2 PRA model for the SAMA analysis were prepared by the CEOG peer review team leaders at Westinghouse, reviewed by the sponsoring utilities (who participated in many of the formal CEOG peer reviews), and issued as formal Westinghouse guideline WCAP-16341-P (Ref. WEST 2005 in ER Section D.11). The guideline was written with a purpose of meeting (or exceeding) the ASME Standard/Regulatory Guide 1.200 capability Category II requirements. The PVNGS Level 2 PRA model used for the SAMA analysis was developed by ERIN Engineering in December 2007 using this Westinghouse guideline, and the final model was independently reviewed by APS.

E. Internal Regulatory Guide (RG) 1.200, Revision 1, self-assessment in September, 2008

Although compliance with Revision 1 of RG 1.200 was not required for the license renewal application, an internal self-assessment was conducted to evaluate the PVNGS PRA model application for SAMA analysis. The RG 1.200 Category II supporting requirements that were not met were assessed for their impact on SAMA application and SAMA results. The assessment concluded that the PVNGS PRA model is suitable for SAMA analysis application.

The un-met RG 1.200 Category II supporting requirements are classified as follows:

- Lack of Internal Flood Model

  Palo Verde structure and design is post 1975. Most of the NRC issues with internal floods were identified and were factored as input into the design. The highly compartmentalized design of the plant reduces the likelihood of flooding affecting more than one train of mitigating equipment. Currently, PVNGS PRA personnel are active within the nuclear industry in the preparation of Internal Flood Modeling guidelines supported by EPRI.

- No Evidence of Cross Comparison of Plant Initiating Events with other Similar Plants

  The Palo Verde CE System 80 design is unique in the nuclear industry. Only Waterford-3 and SONGS-2 and -3 are somewhat similar to Palo Verde's design. Palo Verde's initiating events are consistent with NRC and EPRI publications. There are no remaining open items from the peer review of 1999. Also, no open items from the Combustion Engineering Owners Group cross comparison of PWR initiating events.

- No Credit taken in the LERF Analysis for Post Core Damage Repairs or Human Actions

  The PVNGS LERF analysis did not credit potential equipment repairs. Also, limited credits were taken from human actions. These unmet supporting requirements add to the safety margins in SAMA results

- Insufficient Uncertainty Analyses in the CDF and LERF analyses

  Application of the PRA model in SAMA analysis was conducted at both the mean failure data values and the 95th percentiles. Some SAMAs that screened out as having negative net result were restored to positive net results from the application of the 95th percentiles.

- Documentation Issues

  No numerical deficiencies in the CDF or LERF analyses were identified from this SR deficiency. PVNGS PRA documentation is steadily improving. Palo Verde has recently joined the EPRI Risk and Reliability Documentation Assist (doc.assist) program. This program is aimed at simplifying and improving PRA documentation.

NRC RAI 1.e

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Figure D.2-1 provides the contribution to CDF by 15 initiators as a percentage of the internal events CDF (5.07E-06/yr). Section D.2.2 identifies that the internal events model consists of 28 initiating events. Clarify the difference between initiators and initiating events in this context. In addition, provide in a table the actual numerical value for the internal events CDF contribution for each of the 28 initiators.

APS Response to RAI 1.e

The label "initiator" as used in Figure D.2-1 is meant to identify a PRA model initiating event. In the Maintenance Rule, the term "initiator" is used to identify certain tasks where the affected component or work activity may cause an initiating event to occur.

The pie chart in Figure D.2-1 shows 15 divisions (slices). These 15 divisions actually represent all initiating events by combining together some of the low contributors. If one adds the designated percent contributions, a total of about 100% is obtained. Consistent with Figure D.2-1, Table 1.e-1 below shows the fractional contribution to CDF by all initiators. Some initiators were subdivided into groups (such as IEATWS). This resulted in a total of 34 entries in the table.

Table 1.e-1
PRA Model Revision 15 CDF Breakdown by Fractional Contribution of Initiators

Initiating Event Name	Description	Fractional Contribution to CDF
IELOOP	Loss of Offsite Power	1.96E-01
IEPBA	Loss of ESF Class Bus A	9.23E-02
IEMISC	Unplanned Reactor Trip	7.61E-02
IECPST	Loss of Condensate Pumps	4.89E-02
IEATWS4	ATWS with LOCA and SGTR	4.78E-02
IETT	Turbine Trip	3.72E-02
IESLOCA	Small LOCA	3.23E-02
IEPBB	Loss of ESF Class Bus B	2.22E-02
IEMLOCA	Medium LOCA	1.64E-02
IECONDVAC	Loss of Condenser Vacuum	1.44E-02
IETCW	Loss of Turbine Cooling Water	1.34E-02
IESGTR	Steam Generator Tube Rupture	1.31E-02
IEFWP	Loss of MFW Pumps	7.78E-03
IEATWS5	ATWS and no MFW	4.49E-03
IEATWS2	ATWS with TT	4.09E-03
IEMSIV	Closure of all Main Steam Isolation Valves	4.02E-03
IEPKBM42	Loss of Class 125VDC M42	3.56E-03
IELLOCA	Large LOCA	2.92E-03
IEHPSC-NC-ISL	Tube Failure in High Pressure Seal Cooler	1.92E-03
IEIAS	Loss of Instrument Air	1.85E-03
IEATWS1	ATWS with LOOP and no MFW	1.77E-03
IEPCW	Loss of Plant Cooling Water	1.39E-03
IEPKAM41	Loss of Class 125VDC M41	1.19E-03
IENAB	Loss of Non-class 13.8 kV Bus B	8.15E-04
IENAA	Loss of Non-class 13.8 kV Bus A	7.71E-04
IENCW	Loss of Nuclear Cooling Water	7.39E-04
IEATWS3	ATWS with TT and no MFW	6.06E-04
IEFLB	Feed Water Line Break	5.70E-04
IEPNA	Loss of Class 1E Vital AC Power Train A	3.98E-04
IESLB	Steam Line Break	1.85E-04
IEPNB	Loss of Class 1E Vital AC Power Train B	1.58E-04
IEDCHVAC	Loss DC Equipment Room HVAC	8.98E-05
IEPKCM43	Loss of Class 125VDC M43	4.28E-05
IEPKDM44	Loss of Class 125VDC M44	4.14E-05

NRC RAI 1.f

Provide the following information regarding the Level 1 Probabilistic Risk Assessment (PRA) used for the Severe Accident Mitigation Alternatives (SAMA) analysis:

Section D.1.4 (p. D-4), identifies that static transfer switches for the Vital Alternating Current (AC) on Unit 1 never received the modifications that Units 2 & 3 did, but then claims that the switch failure and human error probability would be the same order of magnitude and thus "there is no material impact resulting from this plant difference." Explain why the human action is as reliable (on same order of magnitude) as the transfer switch. Provide an assessment of the cost and benefits of implementing the modifications on the Unit 1 static transfer switches for the Vital AC.

APS Response to RAI 1.f

Four independent Class 1E, 120V vital instrumentation and control AC power supplies are provided to supply the four channels of the reactor protective and ESF actuation systems. Each vital AC instrumentation and control power supply consists of one inverter rated at 25 kVA, a transfer switch, a backup voltage regulator, and one distribution panel. Normally, each distribution panel is supplied by the inverter. Each inverter is supplied by a separate Class 1E 125 V-DC subsystem. If an inverter is inoperable, its output is outside the acceptable operating range, or it is to be removed from service for maintenance or testing, a backup supply is provided from a separate Class 1E regulated power supply through the transfer switch. Currently, PVNGS Units 2 and 3 have installed automatic static transfer switches and PVNGS Unit 1 utilizes manual transfer switches for each AC power supply.

The estimated cost of replacing the Unit 1 manual transfer switches with automatic static transfer switches on the Class 1E 120V power supplies is $180,000.

The benefit in risk reduction from replacing the Unit 1 Class 1E 120V manual transfer switches with automatic static transfer switches is negligible. This result is based on a marginal difference between the failure probability of an automatic static transfer switch (3.0E-3) and a Human Reliability Assessment (HRA) of the probability an operator fails to perform the transfer of a manual switch (5.6E-3), and the low risk significance of transfer switches in the PRA model. The EPRI HRA Calculator was used to determine the probability of an operator failing to transfer the manual switch. This operator action is proceduralized, practiced, and relatively simple. There are no performance shaping factors involved that diminish the operator's success probability, such as timing, environment, complexity, accessibility or workload.

It should be noted that there is an action in the PVNGS corrective action program to develop and approve a modification to install the static transfer switches in Unit 1 (Condition Report Action Item [CRAI] 3273139).

NRC RAI 2.a

Provide the following information relative to the Level 2 analysis:

Section D.2.2 states that the Level 2 analysis was recently revised to provide a "more realistic treatment of thermal and pressure induced steam generator tube rupture." Explain how the revised treatment is "more realistic." Clarify whether these results have been included in the SAMA analysis.

APS Response to RAI 2.a

Prior to the 2007 revision of the PVNGS PRA, the PVNGS large early release frequency (LERF) analysis for containment failure probabilities was based on the guidance provided in NUREG/CR-6595 which meets Regulatory Guide 1.200, Revisions 1 and 2, Capability Category I. In 2005, APS, with other utilities, sponsored the generation of a Proprietary Westinghouse document (Ref. WEST 2005 in ER Section D.11) to provide a guideline for Level 2 analysis that met or exceeded the more stringent Capability Category II requirements in Regulatory Guide 1.200, Revisions 1 and 2 for PRAs. The new guidelines provided (among other data) plant-specific probabilities for thermal and pressure induced steam generator tube rupture and containment failure probabilities under various accident scenarios, which resulted in a more realistic treatment of these events. These more realistic treatments were incorporated into the 2007 PVNGS Level 2 PRA model and were used for the SAMA analysis.

NRC RAI 2.b

Provide the following information relative to the Level 2 analysis:

Section D.2.2 states that the updated Level 2 model used for the SAMA analysis is also capable of evaluating power uprates. Clarify whether power uprates are currently anticipated and, if so, how power uprates would affect the SAMA analysis.

APS Response to RAI 2.b

There is no plan for power uprate in any of the three PVNGS units.

NRC RAI 2.c

Provide the following information relative to the Level 2 analysis:

Provide a description of the process used to map the Level 1 results into the Level 2 analysis. Describe the plant damage states and how they were applied.

APS Response to RAI 2.c

The implementation of the Level 2 model event trees from Westinghouse document WCAP-16341-P (Ref. WEST 2005 in ER Section D.11) required that new plant damage states (PDSs) be defined for proper quantification in the new Level 2 model event trees. A review of the station blackout (SBO) and non-SBO containment event trees in WCAP-16341-P revealed that the states listed in Table 2.c-1 below would be sufficient to represent unique PDS sets required for quantification. The remaining nodes in the containment events trees would then be set up to use point estimate values and/or system failures as appropriate in the linked event tree/fault tree methodology. This would ensure that proper dependencies were captured between the Level 1 and Level 2 model results. The PDSs and how they were applied are shown in Tables 2.c-1, 2.c-2, and 2.c-3 below.

TABLE 2.c-1:
EVENT TREE GROUPS

PDS	DESCRIPTION	STATION BLACKOUT?	CONTAINMENT BYPASSED?	RCS PRESSURE HIGH?
1A	Non-SBO, BYPASS-ISLOCA	No	Yes	Not Applicable
1B	Non-SBO, BYPASS-SGTR	No	Yes	Not Applicable
2	Non-SBO, RCS @ Low Pressure	No	No	No
3	Non-SBO, RCS @ High Pressure	No	No	Yes
4	SBO, BYPASS	Yes	Yes	Not Applicable
5	SBO, RCS @ Low Pressure	Yes	No	No
6	SBO, RCS @ High Pressure	Yes	No	Yes

In RiskSpectrum software, boundary condition sets were then defined to impose the appropriate boundary conditions for each applicable PDS. The definition of the boundary condition sets and house event settings for each is shown in Table 2.c-2 below.

TABLE 2.c-2
BOUNDARY CONDITION SETS

HOUSE EVENT	BOUNDARY CONDITION SET					
	L2-SBO-BYPASS	L2-SBO-LOW	L2-SBO-HIGH	L2-NOSBO-BYPASS	L2-NOSBO-LOW	L2-NOSBO-HIGH
L2-PDS-SBO	True	True	True	False	False	False
L2-PDS-NO-SBO	False	False	False	True	True	True
L2-BYPASS	True	False	False	True	False	False
L2-NO-BYPASS	False	True	True	False	True	True
L2-RCS-HIGH	False	False	True	False	False	True
L2-RCS-LOW	False	True	False	False	True	False

A review of all Level 1 core damage sequences was then performed to define a representative PDS for each Level 1 sequence. No Level 1 core damage sequences were identified that were applicable to PDS 4 and PDS 5 (because the SBO-SGTR and SBO-ISLOCA cases are not explicitly considered in the Level 1 model). As a result, event trees and boundary condition sets (L2-SBO-BYPASS and L2-SBO-LOW) applicable to PDS 4 and 5 were not applied. The collection of all of the sequences for each PDS is then used as input into the five different applicable Level 2 event trees with the appropriate boundary condition sets applied. The five applicable Level 2 event trees and associated boundary condition set are listed in Table 2.c-3 below.

TABLE 2.c-3:
LEVEL 2 EVENT TREES AND BC SETS

LEVEL 2 EVENT TREE	DESCRIPTION	APPLICABLE BOUNDARY CONDITION SET
L2-PDS1A	Non-SBO, BYPASS-ISLOCA	L2-NOSBO-BYPASS
L2-PDS1B	Non-SBO, BYPASS-SGTR	L2-NOSBO-BYPASS
L2-PDS2	Non-SBO, RCS @ Low Pressure	L2-NOSBO-LOW
L2-PDS3	Non-SBO, RCS @ High Pressure	L2-NOSBO-HIGH
L2-PDS6	SBO, RCS @ High Pressure	L2-SBO-HIGH

NRC RAI 2.d

Provide the following information relative to the Level 2 analysis:

Provide a description of the process used to group the containment event tree (CET) end states into release categories. Identify the number of CETs developed for the Level 2 analysis and describe how they correlate to release categories and plant damage states. Provide a typical CET showing release categories assigned to each end state.

APS Response to RAI 2.d

The PVNGS Level 2 fault tree structure was developed using the containment event tree structure from Westinghouse document WCAP-16341-P (Ref. WEST 2005 in ER Section D.11). The event tree structures are provided below in Figures 2.d-1 and 2.d-2. Figure 2.d-1 shows the non-station blackout (SBO) event tree and Figure 2.d-2 shows the SBO event tree.

Each sequence through the containment event tree results in a unique endstate. Each endstate is from one of four categories: (1) intact, (2) large early release (LERF), (3) small early release (SERF), or (4) late. The endstates are then sequentially numbered.

Figure 2.d-1: Non-Station Blackout Level 2 Event Tree

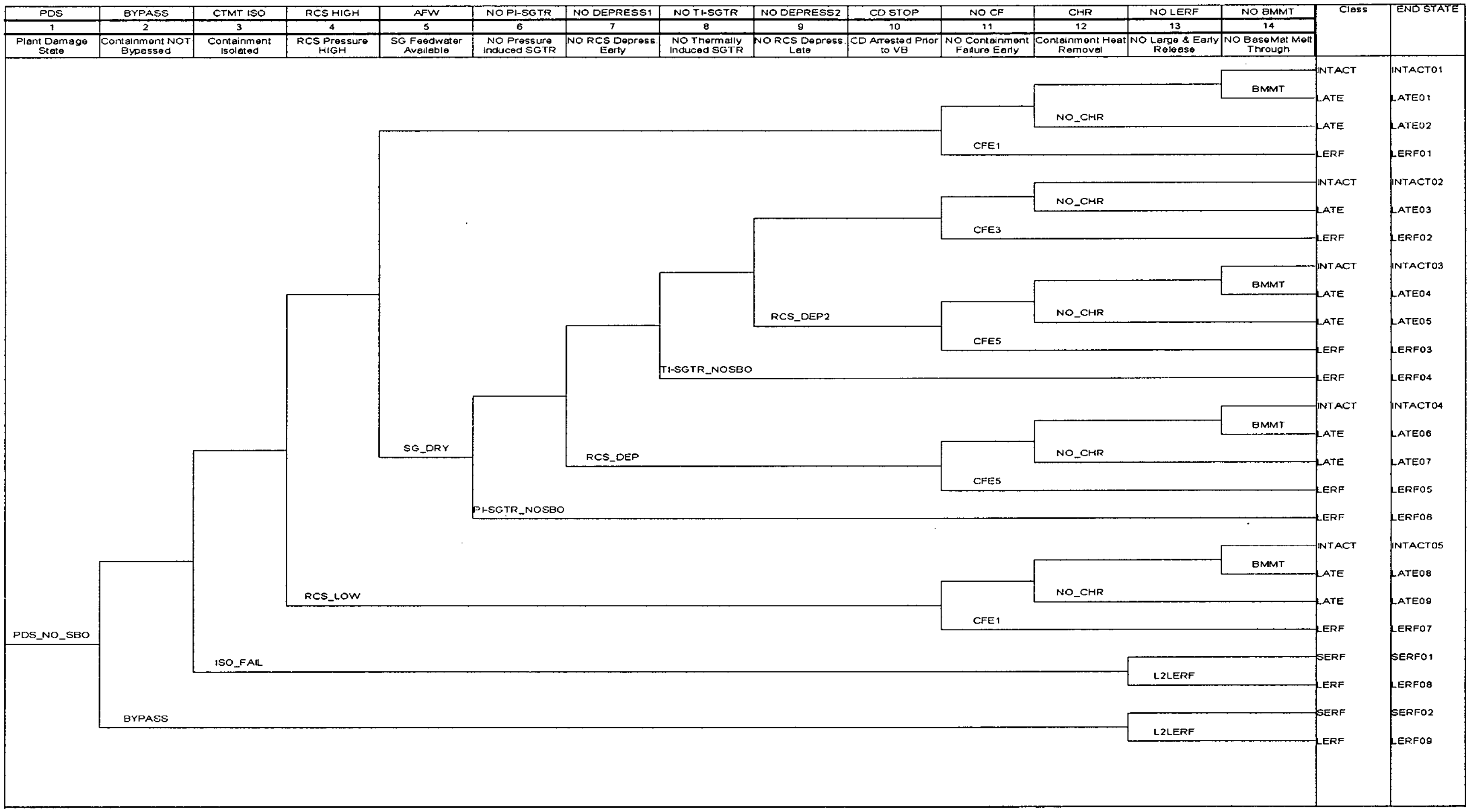
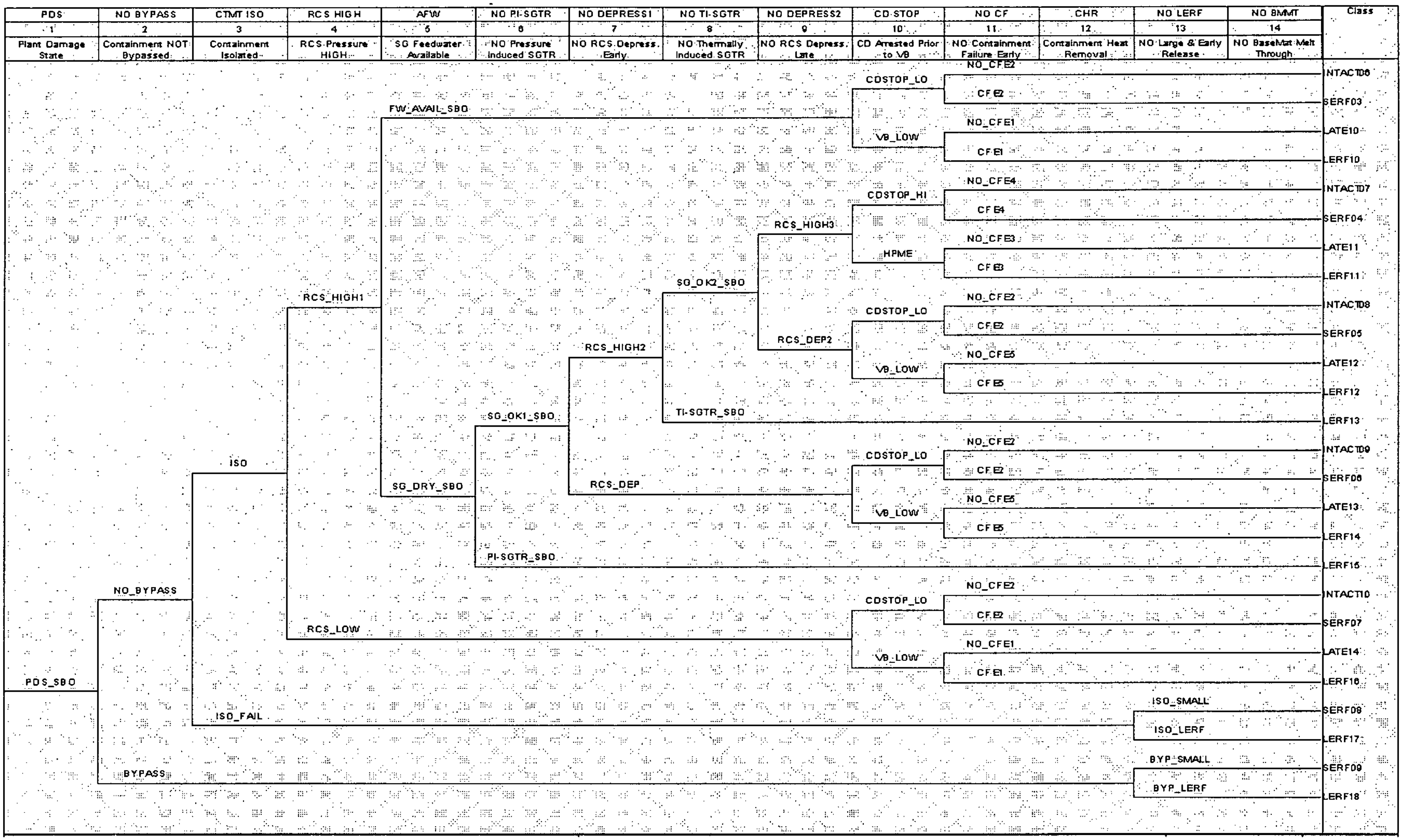


Figure 2.d-2: Station Blackout Level 2 Event Tree



There are 14 nodes or branch points in each non-SBO and SBO containment event tree. The nodes are either split fractions (SF) representing the probability of occurrence of the event, or the node is represented by a fault tree structure containing basic events and human actions. Table 2.d-1 provided below lists the nodes for each containment event tree, its structure and associated probability.

TABLE 2.d-1
CONTAINMENT EVENT TREE NODES

BRANCH	STRUCTURE	PROBABILITY OR GATE	COMMENT
Event Tree L2-PDS1A,L2-PDS1B, L2-PDS2, L2-PDS3			
BYPASS	SF	0 or 1	1.0 (TRUE) for ISLOCA or SGTR sequences, 0.0 (FALSE) for all others
NO_BYPASS	SF	0 or 1	1.0 (TRUE) for all sequences except ISLOCA and SGTR
ISO_FAIL	SF	ISO_FAIL	OR gate of L2-CTMN-ISO-FLR-ILRT, GCTMNT-ISOL-FLR. These are containment leak and containment isolation equipment respectively.
RCS_LOW	SF	0 or 1	Reactor vessel rupture, large LOCA and medium LOCA all result in RCS at low pressure. Additionally, all scenarios with successful RCS depressurization also result in the RCS at low pressure. All other scenarios are assumed high pressure.
RCS_HIGH	SF	0 or 1	Reactor vessel rupture, large LOCA and medium LOCA all result in RCS at low pressure. Additionally, all scenarios with successful RCS depressurization also result in the RCS at low pressure. All other scenarios are assumed high pressure.
SG_DRY	Fault Tree	SG_DRY	OR gate of GAF1, GAF2. This logic checks if feedwater flow is available to both SGs after core damage. This is separately asked since the Level 1 AFW TOP success criteria does not require flow to both SGs.
PI-SGTR_NOSBO	SF	7.88E-03	Based on PI-SGTR event tree described in Appendix C
NO_PI-SGTR_NOSBO	SF	9.92E-01	Based on PI-SGTR event tree described in Appendix C
RCS_DEP	SF	7.5E-02	Based on PSV failure to close early as described in Append D of Ref. 4.
NO_RCS_DEP	SF	9.25E-01	Based on PSV failure to close early as described in Append D of Ref. 4.
TI-SGTR_NOSBO	SF	5.29E-02	Based on TI-SGTR event tree described in Appendix C

TABLE 2.d-1 (cont.)
CONTAINMENT EVENT TREE NODES

BRANCH	STRUCTURE	PROBABILITY OR GATE	COMMENT
NO_TI-SGTR_NOSBO	SF	9.47E-01	Based on TI-SGTR event tree described in Appendix C
RCS_DEP2	SF	9.03E-01	Based on PSV failure to close late or as described in Append D of Ref. 4 and hot leg/surge line failure in Ref. 4.
NO_RCS_DEP2	SF	9.7E-02	Based on PSV failure to close late or as described in Append D of Ref. 4 and hot leg/surge line failure in Ref. 4.
CFE1	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE1	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE2	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE2	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE3	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE3	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE4	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE4	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE5	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE5	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CHR_FAILS	Fault Tree	GCHR	Probability is based on fault tree structure
L2LERF	SF	1.00E+00	No credit taken for source term reductions of bypass scenarios.
NO_L2LERF	SF	0.00E+00	No credit taken for source term reductions of bypass scenarios.
BMMT	SF	1.00E+00	Based on Table 6.14-1 of Ref. 4.
NO_BMMT	SF	0.00E+00	Based on Table 6.14-1 of Ref. 4.
Event Tree L2-PDS6			
BYPASS	SF	0 or 1	1.0 (TRUE) for ISLOCA or SGTR sequences, 0.0 (FALSE) for all others
NO_BYPASS	SF	0 or 1	1.0 (TRUE) for all sequences except ISLOCA and SGTR
ISO_FAIL	SF	ISO_FAIL	OR gate of L2-CTMN-ISO-FLR-ILRT, GCTMNT-ISOL-FLR with BC Set SBO. These are containment leak and containment isolation equipment respectively.
RCS_LOW	SF	0	No SBO low pressure SBO sequences from the Level 1

TABLE 2.d-1 (cont.)
CONTAINMENT EVENT TREE NODES

BRANCH	STRUCTURE	PROBABILITY OR GATE	COMMENT
RCS_HIGH	SF	1	All Level 1 SBO sequences are high pressure.
SG_DRY_SBO	Fault Tree	SG_DRY	OR gate of GAF1, GAF2 with BC Set SBO. This logic checks if feedwater flow is available to both SGs after core damage. This is separately asked since the Level 1 AFW TOP success criteria does not require flow to both SGs.
L2PI-SGTR_SBO	SF	7.88E-03	Based on PI-SGTR event tree described in Appendix C
NO_L2PI-SGTR_SBO	SF	9.92E-01	Based on PI-SGTR event tree described in Appendix C
RCS_DEP	SF	7.5E-02	Based on PSV failure to close early as described in Append D of Ref. 4.
NO_RCS_DEP	SF	9.25E-01	Based on PSV failure to close early as described in Append D of Ref. 4.
L2TI-SGTR_SBO	SF	4.04E-02	Based on TI-SGTR event tree described in Appendix C. The difference from non-SBO case is operator is unable to bump a reactor coolant pump without power available.
NO_L2TI-SGTR_SBO	SF	9.60E-01	Based on TI-SGTR event tree described in Appendix C. The difference from non-SBO case is operator is unable to bump a reactor coolant pump without power available.
RCS_DEP2	SF	9.03E-01	Based on PSV failure to close late or as described in Append D of Ref. 4 and hot leg/surge line failure in Ref. 4.
NO_RCS_DEP2	SF	9.7E-02	Based on PSV failure to close late or as described in Append D of Ref. 4 and hot leg/surge line failure in Ref. 4.
VB_LOW/NO_HPME	SF	1.00E+00	No credit taken for stopping core damage prior to vessel failure.
NO_VB_LOW/HPME	SF	0.00E+00	No credit taken for stopping core damage prior to vessel failure.
CFE1	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE1	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE2	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE2	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE3	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE3	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE4	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.

TABLE 2.d-1 (cont.)
CONTAINMENT EVENT TREE NODES

BRANCH	STRUCTURE	PROBABILITY OR GATE	COMMENT
NO_CFE4	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
CFE5	SF	0.00E+00	Based on Table 6-11.3 of Ref 4.
NO_CFE5	SF	1.00E+00	Based on Table 6-11.3 of Ref 4.
L2LERF	SF	1.00E+00	No credit taken for source term reductions of bypass scenarios.
NO_L2LERF	SF	0.00E+00	No credit taken for source term reductions of bypass scenarios.

If the structure is not a split fraction, it is a fault tree gate structure and the table indicates in the comment field the structure of the gate. For the nodes that are solely split fractions, the probabilities are taken primarily from WCAP-16341-P. For those nodes that are not split fractions, guidance for the fault tree structure is also provided in WCAP-16341-P. The 10 release categories are shown in Table 2.d-2 provided below.

TABLE 2.d-2
REFINED PVNGS LEVEL 2 RELEASE CATEGORIES

RELEASE CATEGORY	DESCRIPTION
INTACT	Containment remains intact including accident sequences that do not lead to containment failure in the long term. The release of fission products (and attendant consequences) is determined from the nominal leakage rate for the plant.
LATE-BMMT-AFW	Late containment failure due to base-mat melt-through with long term AFW available.
LATE-BMMT-NOAFW	Late containment failure due to base-mat melt-through with long term AFW not available.
LATE-CHR-AFW	Late containment failure due to late overpressure with containment heat removal unavailable, but with long term AFW available.
LATE-CHR-NOAFW	Late containment failure due to late overpressure with containment heat removal unavailable, and with long term AFW also unavailable.
LERF-BYPASS	This release category is assigned to that subset of LERF bypass scenarios that result from ISLOCA initiators.
LERF-ISO	This release category includes those sequences that lead to early release due to an undetected pre-existing or subsequent containment isolation failure.

TABLE 2.d-2 (cont.)
REFINED PVNGS LEVEL 2 RELEASE CATEGORIES

RELEASE CATEGORY	DESCRIPTION
LERF-CFE	This release category includes those sequences that result in early containment failure due to severe accident phenomena at or near the time of vessel failure. Based on application of the WCAP methodology in Ref 4. for containment failure probability, PVNGS containment failure resilience documented in Ref.5 and Ref.6 and current Level 2 model assumptions these sequences are currently zero, but are included as a separate category for potential sensitivity study investigation.
LERF-SGTR	This release category is assigned to that subset of LERF bypass scenarios that result from early SGTR scenarios.
SERF	This release category is assigned to all early releases that have the source term reduced from LERF due to some phenomenological means. Based on current Level 2 model assumptions these sequences are currently zero, but are included as a separate category for potential sensitivity study investigation.

NRC RAI 2.e.i

Provide the following information relative to the Level 2 analysis:

The information provided in Sections D.2.7 and D.3.3 does not sufficiently describe how the fission product release fractions were developed for each release category.

Identify and describe the number of Modular Accident Analysis Program (MAAP) calculations made to obtain the fission product release fractions for each release category.

APS Response to RAI 2.e.i

Table 2.e.i-1 shown below (from Table 9 of the PVNGS PRA Level 2 model) identifies and describes the MAAP calculations made to obtain the fission product release fractions. It should be mentioned that the 12 Level 2 cases using MAAP4 to support the SAMA analysis were prepared by ERIN Engineering and independently reviewed by APS. Also, the PVNGS MAAP4 parameter file was prepared and reviewed by industry experts from Fauske & Associates, Inc. Furthermore, an updated initial core inventory was prepared for the analysis.

TABLE 2.e.i-1 KEY EVENT TIMINGS

CASE	NAME	DESCRIPTION	AFW (Y/N)	CS (Y/N)	HLCR (Y/N)	BMMT (Y/N)	SG DRY HOURS	TCU HOURS	TCD HOURS	TVF HOURS	TCF HOURS	TEND HOURS	NG FRACTION	CSI FRACTION
Case 1	INTACT	Trip, No AFW, No RCS depressurization, No induced ruptures, No feed and bleed, VF @ High Press > HPME, 1 CS OK	N	Y	N	N	1.3	1.6	2.1	3.4	NA	48.0	9.4E-04	1.6E-07
Case 2	LATE-BMMT-AFW	Small LOCA, No injection, No RCS depressurization, AFW OK, 1 CS OK	Y	Y	N	Y	NA	0.8	1.1	11.5	34.9	72.0	9.8E-01	3.9E-06
Case 3	LATE-BMMT-NOAFW	Trip, No AFW, No feed and bleed, RCS depressurization via an induced rupture prior to vessel failure, 1 CS OK	N	Y	Y	Y	0.6	1.2	1.7	5.1	21.6	72.0	1.0E+00	1.1E-03
Case 4a	LATE-BMMT-PDS2	Large LOCA, No injection, AFW OK, 1 CS OK	N	Y	N	Y	NA	12 sec	0.5	2.2	16.9	72.0	1.0E+00	3.3E-04
Case 4b	LATE-BMMT-ODS2	Large LOCA, No injection, No AFW, 1 CS OK	N	Y	N	Y	NA	12 sec	0.5	2.2	17.0	72.0	1.0E+00	2.4E-04
Case 5	LATE-CHR-AFW	SBO, AFW OK, RCS depressurized, No injection, No containment spray	Y	N	N	Y	6.3	7.0	7.9	9.5	44.8	72.0	1.0E+00	1.2E-02
Case 6	LATE-CHR-NOAFW	SBO, No AFW, No injection, No containment sprays, induced rupture in the hot leg prior to the time of vessel failure	N	N	Y	Y	1.3	1.6	2.1	5.9	23.1	72.0	1.0E+00	2.3E-01
Case 7	LATE-CHR-PDS2	Small LOCA, AFW OK, RCS depressurized, Initial injection available, but with injection and sprays failed in recirculation mode	Y	Y	N	Y	NA	4.2	5.0	9.3	37.7	72.0	1.0E+00	8.8E-03
Case 8	LERF-BYPASS	ISLOCA with no injection available after RWST depletion	N	N	N	Y	NA	38 sec	0.2	1.7	0.0	72.0	1.0E+00	9.7E-01
Case 9	LERF-ISO	Small LOCA, No AFW, No injection, No containment heat removal, Large pre-existing containment isolation failure	N	N	N	N	1.7	0.8	1.1	3.0	0.0	72.0	1.0E+00	3.3E-01
Case 10	LERF-CFE	Trip, No AFW, No induced ruptures, VF @ High Press > HPME, No containment spray, Containment failure at vessel breach	N	N	N	N	0.6	1.2	1.7	2.8	2.8	72.0	1.0E+00	6.9E-02
Case 11	LERF-SGTR	LOFW, No AFW, No feed and bleed, an induced SGTR and with a SG ADV stuck open	N	N	N	N	1.3	1.6	2.1	3.4	0.0	48.0	8.8E-01	1.6E-01

AFW Aux Feedwater
HLCR Hot Leg Creep Rupture
SG Dry Time of SG Dryout
Tcd Time of core damage (max core > 1800F)
Tcf Time of containment failure
NG Noble Gas release

CS Containment Syray
BMMT Basemat melt thru
Tcu Time of core uncovery
Tvf Time of vessel breach
Tend End Time for MAAP run
CsI CsI release

NRC RAI 2.e.ii

Provide the following information relative to the Level 2 analysis:

The information provided in Sections D.2.7 and D.3.3 does not sufficiently describe how the fission product release fractions were developed for each release category.

Clarify the basis for selecting representative accident sequences.

APS Response to RAI 2.e.ii

The bases for selecting the representative accident sequences were: (1) review of the dominant Level 2 cutsets, and (2) extensive MAAP4 & Level 2 experiences by the preparers. More specifically, Table 2.e.ii-1 provided below shows 12 entries for release categories and the details for each corresponding representative sequence description.

TABLE 2.e.ii-1
PVNGS LEVEL 2 RELEASE CATEGORY REPRESENTATIVE SEQUENCES

RELEASE CATEGORY	SEQUENCE DESCRIPTION
INTACT	Based on the current assumptions in the Level 2 model, the only sequence that can be non-zero is sequence PDS3:0005. This requires that vessel failure occur at high pressure (such that not all of the core debris ends up in the cavity), and that containment heat removal from the containment sprays remains available. Based on a review of the dominant cutsets for this release category, the representative sequence (Case 1) was chosen as a normal trip with no AFW available, no RCS depressurization (including no induced ruptures), no feed and bleed (PVNGS lacks PORVs), but with 1 containment spray available early and in recirc sufficient to avoid late containment failure.
LATE-BMMT-AFW	This release category includes only the PDS3:0002 sequence. This is the only LATE-BMMT case that has long term AFW available. The availability of AFW tends to reduce the overall release of fission products that may occur compared to a similar case without AFW. Based on a review of the dominant cutsets for this release category, the representative sequence (Case 2) was chosen as a small LOCA with common cause failure of the SI injection valves and failure of operators to intiate RCS depressurization. AFW is available and containment heat removal in the form of one containment spray train from the Refueling Water Tank (RWT) and in recirculation is also available. No pre-existing or subsequent containment isolation failures occur and early containment failure does not occur at the time of vessel failure. Late containment failure eventually occurs due to base-mat melt-through (BMMT) since water does not get into the cavity to cool the core debris. Consistent with previous MAAP runs for these types of scenarios at PVNGS, BMMT is assumed to occur when 8' of concrete in the cavity has eroded. A large containment failure area is assumed and retention of fission products in the soil around the base-mat is not credited.

TABLE 2.e.ii-1 (cont.)
PVNGS LEVEL 2 RELEASE CATEGORY REPRESENTATIVE SEQUENCES

RELEASE CATEGORY	SEQUENCE DESCRIPTION
LATE-BMMT-NOAFW	This release category includes sequences PDS3:0009 and PDS3:0014. In all of these cases, AFW is known to have failed, but the RCS is depressurized in the Level 2 analysis via a hot leg surge line rupture or a stuck open PSV. Based on a review of the dominant cutsets for this release category, the representative sequence (Case 3) was chosen as a normal trip with no AFW available, RCS depressurization via an induced rupture prior to vessel failure, no feed and bleed (PVNGS lacks PORVs), but with 1 containment spray available early and in recirc. Similar to Case 2, late containment failure eventually occurs due to base-mat melt-through (BMMT) since water does not get into the cavity to cool the core debris.
LATE-BMMT-PDS2	This release category includes sequence PDS2:0019. These are all low pressure sequences with core debris in the cavity, succesful containment heat removal, no early containment failures, but with eventual base-mat melt-through assumed. Based on a review of dominant cutset for this release category, the representative sequence (Case 4a) was chosen as a Large LOCA with no injection available, but with 1 containment spray train available early and in recirc. Similar to Cases 2 and 3, late containment failure eventually occurs due to base-mat melt-through (BMMT) since water does not get into the cavity to cool the core debris. Since the availability of AFW is not specifically questoined, Case 4a was developed assuming it is available (likely based on a review of cutsets). However, Case 4b was developed to consider the situation where AFW is failed to see if this has a measurable impact on the source term results.
LATE-CHR-AFW	This release category includes sequences PDS3:0003 and PDS6:0003. In these cases compared to the BMMT-AFW scenarios, the main difference is that containment sprays are also failed. A review of the cutsets showed that contributors included various SBO events. The representative sequence (Case 5) was chosen to be an SBO with no injection or containment sprays available. Consistent with the Level 1 evaluation, AFW is available for three hours and is assumed to be failed upon battery depletion at that time if offsite power is not recovered.
LATE-CHR-NOAFW	This release category includes non-SBO sequences PDS3:0006, PDS3:0010, and PDS3:0015 as well as SBO sequences PDS6:0007, PDS6:0011, and PDS6:0016. Based on a review of dominant cutsets, the representative sequence (Case 6) was chosen to be an SBO scenario with no injection, no AFW, and no containment sprays available, but with an induced rupture in the hot leg or surge line prior to the time of vessel failure.

TABLE 2.e.ii-1 (cont.)
PVNGS LEVEL 2 RELEASE CATEGORY REPRESENTATIVE SEQUENCES

RELEASE CATEGORY	SEQUENCE DESCRIPTION
LATE-CHR-PDS2	This release category includes only the PDS2:0020 sequence. It includes sequences with the RCS known to be at low pressure, no early containment failures, but with long term containment heat removal failed. Based on a review of dominant cutsets, the representative sequence (Case 7) was chosen as a small LOCA scenario with AFW available with successful RCS depressurization, with initial injection available, but with injection and sprays failed in recirculation mode.
LERF-BYPASS	This release category encompasses all ISLOCA scenarios and is represented by sequence PDS1A:0025 Based on a review of the dominant contributors, the representative sequence (Case 8) was chosen as an ISLOCA in the hot leg (IEHPSC-NC-ISL) with no injection available. Early core damage occurs with fission product release assumed to be directly to the environment.
LERF-ISO	This release category encompasses all of the containment isolation failure sequences (i.e. PDS2:0023, PDS3:0023 and PDS6:0024). A review of the cutsets showed that numerous intiators contributed. A representative scenario of a small LOCA initiator with no AFW, no injection, and no containment heat removal, but with a large pre-existing containment isolation failure was chosen as a bounding representative sequence (Case 9).
LERF-CFE	This release category includes all sequences that result in early containment failure due to severe accident phenomena at or near the time of vessel failure. This includes sequences PDS2:0021, PDS3:0004, PDS3:0007, PDS3:0011, PDS3:0016, PDS6:0004, PDS6:0008, PDS6:0012, and PDS6:0017. Based on the current Level 2 model assumptions, these sequences all quantify to zero, but in any event this release category is included for completeness. The modeling assumption resulting in the zero value quantification is based on zero probability of containment failure mechanisms for the Palo Verde containments in Tables 6.11-1 through Table 6.11-3 in the WCAP. The representative sequence (Case 10) was chosen as a LOFW with no AFW available, no induced ruptures, and no RCS depressurization. This results in vessel failure at high pressure at which a large containment failure is assumed to occur.
LERF-SGTR	This release category encompasses the steam generator tube rupture scenarios (PDS1B:0025) as well as those scenarios that are predicted to have thermally or pressure induced tube ruptures (PDS3:0012, PDS3:0017, PDS6:0013, and PDS6:0018). The representative sequence (Case 11) was chosen as a total loss of feedwater event with no RCS depressurization, no feed and bleed (PVNGS lacks PORVs), an induced SGTR and with a SG PORV stuck open.
SERF	This release category encompasses all SERF sequences that have the source term reduced from LERF due to some phenomenological means. Based on current Level 2 model assumptions, credit for such source term reduction is not taken and these sequences all quantify to zero. A representative case was therefore not chosen.

NRC RAI 2.e.iii

Provide the following information relative to the Level 2 analysis:

The information provided in Sections D.2.7 and D.3.3 does not sufficiently describe how the fission product release fractions were developed for each release category.

Describe how the release fractions obtained from the MAAP calculations were used to develop release fractions for each CET sequence.

APS Response to RAI 2.e.iii

A 12x12 matrix was generated in the PVNGS Level 2 PRA model to break down the release fraction from each of the 12 release categories by each of the 12 fission product groups. Each of the resulting 144 entries shows the corresponding MAAP4 run and the individual plume release fraction. This matrix is provided in Table 2.e.iii-1 below.

Table 2.E.III-1
Source Term Magnitude and Timing Results

RELEASE CATEGORY	RELEASE CATEGORY[1]											
	INTACT	LATE-BMMT-AFW	LATE-BMMT-NOAFW	LATE-BMMT-PDS2	LATE-BMMT-PDS2-NOAFW	LATE-CHR-AFW	LATE-CHR-NOAFW	LATE-CHR-PDS2	LERF-BYPASS	LERF-ISO	LERF-CFE	LERF-SGTR
Bin Frequency												
MAAP Run	Case 1	Case 2	Case3	Case4a	Case4b	Case5	Case6	Case7	Case8	Case9	Case10	Case11
Run Duration	48 hr	72 hr	72 hr	72 hr	72 hr	72 hr	72 hr	72 hr	72 hr	72 hr	72 hr	48 hr
Time after Scram when General Emergency is declared (3)	2.1 hr	1.1 hr	1.7 hr	.5 hr	.5 hr	7.9 hr	2.1 hr	5.0 hr	.2 hr	1.1 hr	1.7 hr	2.1 hr
Fission Product Group:												
1) Noble												
Total Plume 1 Release Fraction	9.40E-04	7.00E-01	6.40E-01	6.60E-01	6.60E-01	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	6.00E-01
Start of Plume 1 Release (hr)	2.00	34.90	21.60	16.90	17.00	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	48.00	36.00	22.60	17.90	18.00	46.80	25.10	40.00	1.20	7.00	5.00	3.10
Total Plume 2 Release Fraction[2]		1.00E+00	1.00E+00	1.00E+00	1.00E+00							8.80E-01
Start of Plume 2 Release (hr)		36.00	22.60	17.90	18.00							36.00
End of Plume 2 Release (hr)		72.00	40.00	40.00	40.00							48.00
2) CsI												
Total Plume 1 Release Fraction	1.60E-07	1.00E-06	2.00E-04	2.00E-04	1.50E-04	7.00E-03	2.30E-01	8.80E-03	9.70E-01	1.80E-01	5.00E-02	1.30E-01
Start of Plume 1 Release (hr)	2.00	1.10	21.60	16.90	17.00	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	10.00	22.60	17.90	18.00	46.80	72.00	72.00	1.20	7.00	5.00	4.00
Total Plume 2 Release Fraction[2]		4.00E-06	1.10E-03	3.30E-04	2.40E-04	1.20E-02				3.30E-01	6.90E-02	1.60E-01
Start of Plume 2 Release (hr)		34.90	22.60	17.90	18.00	46.80				7.00	5.00	4.00
End of Plume 2 Release (hr)		40.00	72.00	72.00	72.00	72.00				72.00	72.00	48.00
3) TeO2												
Total Plume 1 Release Fraction	1.60E-07	8.70E-07	1.90E-05	1.60E-05	1.30E-05	1.20E-03	3.70E-03	8.80E-04	9.60E-01	2.00E-01	5.60E-02	1.50E-01
Start of Plume 1 Release (hr)	2.00	1.10	2.00	1.00	1.00	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	10.00	72.00	72.00	72.00	72.00	72.00	50.00	1.20	7.00	5.00	4.00
Total Plume 2 Release Fraction[2]												
Start of Plume 2 Release (hr)												
End of Plume 2 Release (hr)												

Table 2.E.III-1 (cont.)
Source Term Magnitude and Timing Results

RELEASE CATEGORY	RELEASE CATEGORY[1]											
	INTACT	LATE-BMMT-AFW	LATE-BMMT-NOAFW	LATE-BMMT-PDS2	LATE-BMMT-PDS2-NOAFW	LATE-CHR-AFW	LATE-CHR-NOAFW	LATE-CHR-PDS2	LERF-BYPASS	LERF-ISO	LERF-CFE	LERF-SGTR
4) SrO												
Total Plume 1 Release Fraction	1.40E-08	3.00E-07	9.00E-08	2.00E-07	2.00E-07	4.70E-06	1.40E-05	5.30E-04	7.60E-02	1.70E-02	6.80E-03	7.60E-04
Start of Plume 1 Release (hr)	3.00	4.00	5.10	0.50	0.50	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	5.10	1.50	1.50	46.80	25.10	40.00	1.20	7.00	10.00	4.00
Total Plume 2 Release Fraction[2]												
Start of Plume 2 Release (hr)												
End of Plume 2 Release (hr)												
5) MoO2												
Total Plume 1 Release Fraction	3.10E-08	6.00E-07	1.00E-07	2.30E-07	3.20E-07	5.00E-07	1.40E-05	1.70E-04	6.30E-02	1.30E-01	8.30E-03	3.60E-02
Start of Plume 1 Release (hr)	2.00	4.00	2.00	0.50	0.50	10.00	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	2.00	1.50	1.50	20.00	25.10	40.00	1.20	7.00	5.00	4.00
Total Plume 2 Release Fraction[2]						1.00E-06						
Start of Plume 2 Release (hr)						44.80						
End of Plume 2 Release (hr)						72.00						
6) CsOH												
Total Plume 1 Release Fraction	1.20E-07	1.00E-06	4.00E-05	1.00E-04	1.00E-04	2.00E-03	3.60E-02	2.40E-03	9.70E-01	1.90E-01	3.90E-02	9.00E-02
Start of Plume 1 Release (hr)	2.00	4.00	21.60	16.90	16.90	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	22.60	20.00	20.00	46.80	72.00	72.00	1.20	7.00	7.00	4.00
Total Plume 2 Release Fraction[2]		1.80E-06	1.80E-04			3.20E-03						
Start of Plume 2 Release (hr)		34.90	22.60			46.80						
End of Plume 2 Release (hr)		72.00	72.00			72.00						

Table 2.E.III-1 (cont.)
Source Term Magnitude and Timing Results

RELEASE CATEGORY	RELEASE CATEGORY[1]											
	INTACT	LATE-BMMT-AFW	LATE-BMMT-NOAFW	LATE-BMMT-PDS2	LATE-BMMT-PDS2-NOAFW	LATE-CHR-AFW	LATE-CHR-NOAFW	LATE-CHR-PDS2	LERF-BYPASS	LERF-ISO	LERF-CFE	LERF-SGTR
7) BaO												
Total Plume 1 Release Fraction	2.10E-08	6.00E-07	2.60E-07	3.00E-07	3.00E-07	9.00E-06	2.50E-05	4.90E-04	7.50E-02	4.00E-02	6.90E-03	1.60E-02
Start of Plume 1 Release (hr)	2.00	4.00	2.00	0.50	0.50	44.60	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	72.00	1.50	1.50	72.00	25.10	40.00	1.20	7.00	10.00	4.00
Total Plume 2 Release Fraction[2]												
Start of Plume 2 Release (hr)												
End of Plume 2 Release (hr)												
8) La2O3												
Total Plume 1 Release Fraction	1.40E-08	1.60E-07	5.00E-09	1.00E-08	1.50E-08	2.50E-07	1.30E-06	2.80E-05	2.90E-03	1.20E-02	3.00E-03	1.70E-04
Start of Plume 1 Release (hr)	3.00	12.00	5.00	0.50	2.00	10.00	23.10	37.70	2.00	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	5.00	1.50	3.00	15.00	25.10	40.00	3.00	7.00	7.00	4.00
Total Plume 2 Release Fraction[2]						5.40E-07						
Start of Plume 2 Release (hr)						44.80						
End of Plume 2 Release (hr)						46.80						
9) CeO2												
Total Plume 1 Release Fraction	1.40E-08	2.20E-07	9.00E-08	1.50E-07	1.90E-07	4.30E-06	2.30E-05	6.70E-05	3.90E-02	1.50E-02	7.00E-03	4.40E-04
Start of Plume 1 Release (hr)	3.00	12.00	5.00	0.50	2.00	44.80	23.10	37.70	2.00	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	5.00	1.50	3.00	46.80	25.10	40.00	3.00	7.00	10.00	4.00
Total Plume 2 Release Fraction[2]												
Start of Plume 2 Release (hr)												
End of Plume 2 Release (hr)												

Table 2.E.III-1 (cont.)
Source Term Magnitude and Timing Results

RELEASE CATEGORY	RELEASE CATEGORY[1]											
	INTACT	LATE-BMMT-AFW	LATE-BMMT-NOAFW	LATE-BMMT-PDS2	LATE-BMMT-PDS2-NOAFW	LATE-CHR-AFW	LATE-CHR-NOAFW	LATE-CHR-PDS2	LERF-BYPASS	LERF-ISO	LERF-CFE	LERF-SGTR
10) Sb												
Total Plume 1 Release Fraction	2.00E-07	1.00E-06	4.00E-05	3.00E-05	2.50E-04	6.10E-02	1.70E-02	4.60E-03	5.00E-01	3.40E-01	1.70E-01	1.00E-01
Start of Plume 1 Release (hr)	3.00	4.00	21.60	16.90	17.00	44.80	23.10	37.70	0.20	1.10	2.80	2.10
End of Plume 1 Release (hr)	4.00	12.00	22.60	17.90	18.00	46.80	72.00	50.00	1.20	7.00	10.00	4.00
Total Plume 2 Release Fraction[2]		1.10E-05	7.20E-05	5.00E-05								2.10E-01
Start of Plume 2 Release (hr)		34.90	22.60	17.90								4.00
End of Plume 2 Release (hr)		72.00	72.00	72.00								48.00
11) Te2												
Total Plume 1 Release Fraction	3.00E-10	1.70E-05	3.00E-09	9.00E-07	4.60E-07	2.50E-04	1.00E-03	1.70E-03	3.10E-03	1.20E-03	7.30E-04	2.30E-04
Start of Plume 1 Release (hr)	3.00	34.90	21.60	16.90	17.00	44.80	23.10	37.70	2.00	3.00	2.80	36.00
End of Plume 1 Release (hr)	4.00	72.00	22.60	17.90	18.00	46.80	72.00	72.00	3.00	72.00	10.00	48.00
Total Plume 2 Release Fraction[2]			6.50E-09			3.40E-04						
Start of Plume 2 Release (hr)			22.60			46.80						
End of Plume 2 Release (hr)			72.00			72.00						
12) UO2												
Total Plume 1 Release Fraction	0.00E+00	1.50E-09	2.70E-09	3.00E-09	3.00E-09	1.00E-07	2.50E-07	1.00E-07	1.70E-04	3.30E-06	9.70E-06	5.20E-08
Start of Plume 1 Release (hr)		34.90	5.00	16.90	17.00	44.80	23.10	37.70	2.00	7.00	7.00	10.00
End of Plume 1 Release (hr)		72.00	72.00	17.90	18.00	72.00	25.10	50.00	3.00	10.00	10.00	48.00
Total Plume 2 Release Fraction[2]												
Start of Plume 2 Release (hr)												
End of Plume 2 Release (hr)												

(1) Puff releases are denoted in the table by those entries with equivalent start and end times.
(2) Plume 2 release fraction is cumulative and includes the initial plume 1 release fraction.
(3) General Emergency declaration assumed to be time of core damage.

NRC RAI 2.e.iv

Provide the following information relative to the Level 2 analysis:

The information provided in Sections D.2.7 and D.3.3 does not sufficiently describe how the fission product release fractions were developed for each release category.

Clarify whether the MAAP calculations were performed before or after the Level 2 update and identify the version of the MAAP code used in the SAMA evaluation.

APS Response to RAI 2.e.iv

The MAAP4 calculations were performed as part of the PVNGS PRA Level 2 update before the SAMA analysis was performed. Also, the SAMA analysis was foreseen as an upcoming major application of the new Level 2 model. MAAP code revision 4.0.5 was used in the analysis.

NRC RAI 3.a.i

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Section D.5.1.6.1 provides the fire CDF (Total 2.72E-06/yr) for the top 10 contributing fire compartments. The reported values are substantially reduced from those reported in the Individual Plant Examination of External Events (IPEEE) (i.e., a total fire CDF of 8.67E-05 per year).

Provide a description of the fire PRA development since the IPEEE. Identify the model changes that most impacted the reduction in fire CDF.

APS Response to RAI 3.a.i

See the APS responses to RAIs 3.a.ii and 3.a.iii below.

NRC RAI 3.a.ii

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Section D.5.1.6.1 provides the fire CDF (Total 2.72E-06/yr) for the top 10 contributing fire compartments. The reported values are substantially reduced from those reported in the Individual Plant Examination of External Events (IPEEE) (i.e., a total fire CDF of 8.67E-05 per year).

Provide a description of the quality controls applied to the development of the fire PRA model. Identify and discuss any internal and external reviews since the 2003 fire PRA peer review. Describe any significant review comments, their resolution, and the potential impact of any unresolved comments on the results of the SAMA analysis.

APS Response to RAI 3.a.ii

The EPRI fire events database (NSAC/178L Rev. 1) was used, along with the EPRI fire PRA methodology (TR-105928). Data and model development were guided by an outside consultant. All work was subject to applicable qualification and independent verification requirements per PVNGS administrative controls. A peer review was performed by ERIN Engineering in 2003. Several facts and observations (F&Os) were generated; most have been resolved. There are five open Category C and D F&Os, and no open Category A or B F&Os. Three of the remaining open F&Os are primarily documentation issues that will have little or no impact on results. The other two deal with the need to update initiating event frequencies and treatment of fire-induced circuit failures. The treatment of both of these issues is under development by industry and NRC in relation to fire PRA standards and application to NFPA-805.

The following Category A and B F&Os have been addressed:

- Inconsistency was noted in how fire-wrapped conduit was treated. The resolution made clarifications in documentation only. No model changes were necessary.

- A suggestion was made to examine HEPs to ensure that the actions could be carried out from an accessibility perspective. No problems were found.

- An error in the initiating event frequency partitioning for the Train B ESF (Essential Safety Features) Switchgear Room was noted. This was corrected resulting in a 16% increase in fire CDF.

- A suggestion was made to re-examine the basis for assigning probabilities of manual reactor trip (as a function of fire severity). This resulted in no changes to the model.

- Credit for CO_2 actuation in the switchgear rooms was questioned with regard to thermal sensing devices. Evaluation resulted in credit for suppression being removed, which resulted in a 14% increase in fire CDF.

- One Control Room fire sequence was non-conservative regarding the potential impact to off-site power. The sequence was corrected; there was no measureable increase in fire CDF.

NRC RAI 3.a.iii

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Section D.5.1.6.1 provides the fire CDF (Total 2.72E-06/yr) for the top 10 contributing fire compartments. The reported values are substantially reduced from those reported in the Individual Plant Examination of External Events (IPEEE) (i.e., a total fire CDF of 8.67E-05 per year).

For each of the dominant fire compartments, explain what measures, if any, have already been taken (since the IPEEE) and credited in the fire PRA to reduce fire risk. Include in the response specific improvements to fire detection systems, enhancements to fire suppression capabilities, changes that would improve cable separation, and improvements to processes/procedures for monitoring and controlling the quantity of combustible materials in critical areas.

APS Response to RAI 3.a.iii

The IPEEE fire PRA model was substantially updated and incorporated into the living at-power PRA model in Revision 7 during 2001. Substantial reductions in the IPEEE fire CDF and LERF were realized during this revision of the fire PRA model. The revisions to the fire PRA model are too numerous to list. Due to the number of changes made from the IPEEE fire PRA model, the updated fire PRA model was subsequently peer reviewed by ERIN Engineering in 2003. The results of the peer review indicated that the Fire PRA for PVNGS is complete and comprehensive. All level A and B facts and observations (F&Os) from the ERIN peer review were subsequently resolved in later revisions of the fire PRA model.

No physical changes to the plant have been done either as a result of the IPEEE (Fire-Induced Vulnerability Evaluation) or the later fire PRA. However, as a result of the IPEEE, the Corridor Building was added to the combustible material control program. The corridor building contains off-site power bus ducts and off-site power circuit breaker control cables.

NRC RAI 3.a.iv

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Section D.5.1.6.1 provides the fire CDF (Total 2.72E-06/yr) for the top 10 contributing fire compartments. The reported values are substantially reduced from those reported in the Individual Plant Examination of External Events (IPEEE) (i.e., a total fire CDF of 8.67E-05 per year).

The table on page D-39 indicates that one of the reasons the PVNGS internal fire model is overly conservative is that it is more detailed than other models. The main motivation for increasing the level of detail in failure models is to reduce the conservatism (e.g., make the model more realistic). Explain how more detail in the PVNGS fire model introduces more conservatism.

APS Response to RAI 3.a.iv

The text that introduces the table on LRA page D-39 implies that it summarizes the features of the fire modeling process that make it undesirable to directly compare the fire and internal event CDFs. However, the table was actually used to provide a summary of the characteristics of the PVNGS fire model whether they were conservative or not. The table was not intended to suggest that the increased level of detail in the fire model was a source of conservatism.

The table on LRA page D-39 is clarified as shown below by adding the following text to the entries for "Sequences" and "Level of Detail": "No significant sources of conservatism identified for PVNGS." The revised paragraphs, included in LRA Amendment No. 2 provided in Enclosure 2, are as follows:

Sequences:	No significant sources of conservatism were identified for PVNGS. Sequences in the PVNGS fire model are defined in detail. The consequences of any sequence collapsing is likely minor.
Level of Detail:	No significant sources of conservatism were identified for PVNGS. Many fire PRAs may have reduced level of detail in the mitigation of the initiating event and consequential system damage; however, the PVNGS model includes a detailed assessment of the impacts of the initiating events, consequential fire damage, and the subsequent response of the plant.

NRC RAI 3.b

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

The SAMA analysis assumes that risks posed by external and internal events is approximately equal (page D-54). Based on this assumption, the estimated benefit from reduction of internal event risk was doubled to account for a corresponding reduction in external event risk (with the exception that fire risk was removed from the external event multiplier and calculated separately). However, page D-54 estimates the CDF from external events to be 6.72E-06 per year, a factor of 1.3 greater than the internal events CDF (5.07E-06 per year) used in the SAMA analysis. Furthermore, in "Request for Amendment to Technical Specification 5.5.16, Containment Leakage Rate Testing

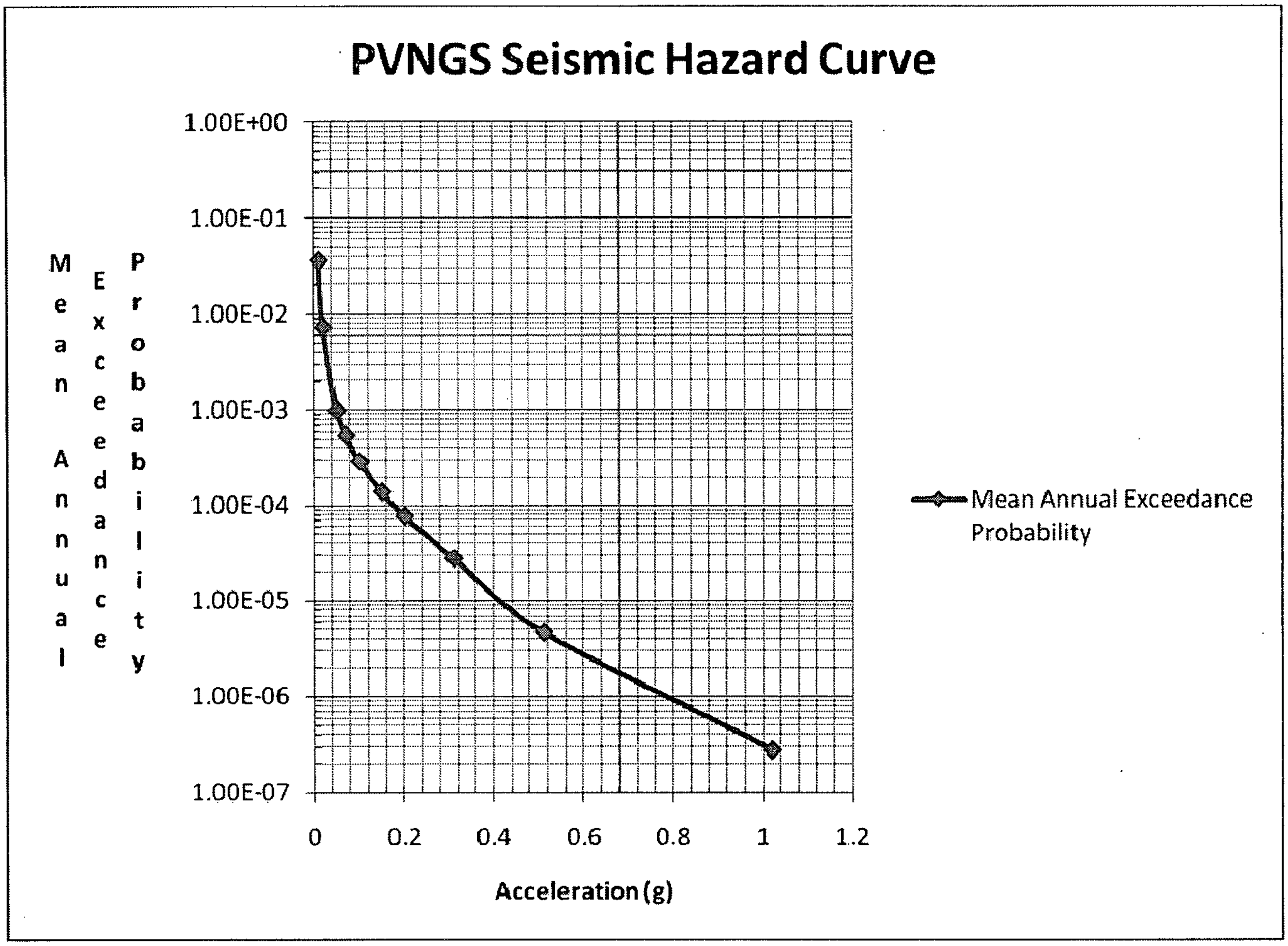
Program" (Arizona Public Service Company [APS] Letter to NRC #102-05902-JHH/DFS, dated October 1, 2008), APS estimated the seismic CDF for PVNGS to be about 7.49E-06 per year using the approximation method described in a paper by Robert P. Kennedy, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations," and using seismic hazard curve data for Palo Verde. Based on this, provide justification for why a multiplier of 3.6 [(5.72E-06 + 7.49E-06) / 5.07E-06 + 1] shouldn't be used to account for the additional risk of all external events (seismic, fire, high winds, etc.) rather than the multiplier of two used in the SAMA analysis.

APS Response to RAI 3.b

The methodology generated by Dr. Robert P. Kennedy for estimating a plant seismic CDF from an existing seismic margins analysis was applied conservatively in APS letter no. 102-05902-JHH/DFS to support the integrated containment leakage rate testing (ILRT) program amendment request. Conservatism (beyond that built into Dr. Kennedy's methodology) in the seismic CDF estimate for ILRT extension application was acceptable because the results and conclusions from that analysis were not sensitive to slight variations in the external events CDF. For the SAMA analysis, however, the external events multiplier and the analysis results are very sensitive to slight variations in seismic CDF. Hence, it was appropriate to review the various steps in the calculation to determine if the process could be further refined for a best estimate seismic CDF.

Upon review, it was concluded that the method used to estimate the "hazard exceedance frequency" ($H_{10\%}$) in APS letter no. 102-05902-JHH/DFS was slightly conservative. Specifically, the use of linear interpolation to estimate $H_{10\%}$ for a non-linear curve resulted in an increased value for $H_{10\%}$. When the available data points for the Palo Verde seismic hazard curve are plotted, as shown in Figure 3.b-1 below, it can be seen that a value of 9.50E-06 for the 0.42g acceleration is a more realistic fit to the existing data than the value of 1.50E-05 shown in Table B-1 on page B-3 in APS letter no. 102-05902-JHH/DFS.

Figure 3.b-1



When 9.50E-06 is used in place of 1.50E-05, the seismic CDF is reduced to 4.75E-06/yr:

$$CDF_{seismic} = 0.5 * H_{10\%} = 0.5 * 9.5E\text{-}06 = 4.75E\text{-}06/yr$$

Having established a revised seismic CDF, the contributions of the other external events must also be reviewed. In section D.4.6.2 of the ER, the external events contributions from the ER were summarized as follows:

IPEEE Contributor Summary

External Event Initiator Group	CDF
Seismic	Not Applicable (seismic margins analysis performed)
Internal Fire (current model)	2.72E-06/yr
High Winds	4.10E-10/yr (quantitative screening information used to develop a CDF for SAMA; refer to section D.5.1.6.3.)
External Floods	Not Applicable (progressive screening method used)
Accidental Aircraft	< 3.00E-08/yr Impact (refer to section D.5.1.6.5)
Others	Not Applicable (progressive screening method used)
Total (for initiators with CDF available)	2.75E-06/yr

Subsequently, CDF estimates of 1.0E-06/yr were assigned to the contributors for which CDF had not been derived in the IPEEE. While these CDF assignments could be made, other values could also be used. Given that these types of events were determined to not pose significant threats to the plant, it is suggested that the accidental aircraft impact CDF of 3.0E-08/yr be used to represent these negligible risks. When the revised values are used, the total external events CDF is only 7.56E-06/yr:

Revised External Events Contributor Summary

External Event Initiator Group	CDF
Seismic	4.75E-06/yr
Internal Fire (current model)	2.72E-06/yr
High Winds	4.10E-10/yr
External Floods	3.0E-08/yr
Accidental Aircraft	3.0E-08/yr
Others	3.0E-08/yr
Total	7.56E-06/yr

The external events CDF of 7.56E-06/yr is a factor of 1.5 greater than the internal events CDF of 5.07E-06/yr (7.56E-06 / 5.07E-06 = 1.5). While these numbers are not exactly equal, it is still considered reasonable to assume that the risk due to both internal and external events is "approximately equal." However, if the revised external events CDF was used to re-calculate the external events multiplier, it would be 2.5 [(5.07E-06 + 7.56E-06) / 5.07E-06 = 2.5].

NRC RAI 3.c

Provide the following information with regard to the treatment and inclusion of external events in the SAMA analysis:

Provide an assessment of the impact on the initial and final SAMA screenings if the internal events benefits are increased by a factor of 3.6, or a revised multiplier developed by APS based on the more realistic estimate of seismic CDF. Provide a Phase II analysis for any Phase I SAMAs that were screened out in the Environmental Report (ER) but would not have been screened out using the revised multiplier.

APS Response to RAI 3.c

As documented in the response to RAI question 3.b, the modified external events multiplier for PVNGS is 2.5. Use of this multiplier has the potential to impact both the Phase I and Phase II SAMA analyses, which are addressed below.

Phase I Re-Analysis

The use of an external events multiplier of 2.5 in place of the multiplier of 2.0 will result in a maximum averted cost-risk (MACR) that is higher than what was used in the ER. As a result, it is necessary to re-perform the Phase I screening process using the updated MACR. Because the SAMA analysis considers the use of the 95th percentile PRA results, it necessary for the Phase I screening process to account for this.

The updated MACR can be calculated for PVNGS by multiplying the original MACR by the ratio of the updated external events multiplier to the original external events multiplier:

$$\$4,668,000 * 2.5 / 2.0 = \$5,835,000$$

In order to account for the 95th percentile PRA results, the updated MACR is multiplied by a factor of 2.7, as described in Section D.7.1.1 of the ER:

$$MACR_{95thpercentile} = 5,835,000 * 2.7 = \$15,754,500$$

In the original Phase I analysis, the following SAMAs were screened based on their costs of implementation exceeding $4,668,000: SAMAs 1, 2, 3, 5, 7, 9, 12, 14, 16, and 18. When the 95th percentile MACR of $15,754,500 is considered, only SAMAs 1, 16, and 18 can still be screened on cost alone. SAMAs 2, 3, 5, 7, 9, 12 and 14 require further evaluation as part of the Phase II re-analysis.

Phase II Re-Analysis

Not all Phase II SAMAs require detailed PRA model quantifications. In some cases, PRA insights can be used to bound the benefits of a SAMA so that detailed modeling does not have to be performed. Of the SAMAs that could no longer be screened in the Phase I analysis due to the increased external events multiplier combined with the 95th percentile PRA results, SAMAs 5, 12 and 14 have been designated for Phase II re-evaluation in conjunction with the original Phase II SAMAs. The remaining SAMAs (2, 3, 7, and 9) have been identified as candidates for screening based on PRA insights due to their high costs of implementation and lack of existing detailed quantifications. The screening based on the PRA insights is addressed first followed by the results of the updated detailed analyses.

Phase II Screening Using PRA Insights

The high costs of implementation for SAMAs 2, 3, 7, and 9 indicate that they could not be cost effective unless the SAMAs eliminate a very large part of plant risk. For each of these SAMAs, the PRA results were reviewed to identify portions of plant risk that could not be impacted by the SAMA (both the CDF and Level 2 composite cutsets were reviewed). Once the limitations on the scenarios that the SAMAs could impact were identified, the maximum percentage in risk reduction could be estimated for each SAMA. The methodology from Section D.4 of the ER was then used to convert the percent reduction in CDF, dose-risk, and off-site economic cost-risk (OECR) to a bounding averted cost-risk.

Because the Level 2 composite frequency is based on only the largest contributors to the Level 3 results, it was assumed that a reduction in the Level 2 composite frequency was proportional to the reduction in dose-risk and OECR. For SAMA 3, the complexity of the SAMA's impacts made it difficult to parse the Level 2 cutsets. As an alternative, the impact of the SAMA was bounded by retaining only the Level 3 results from the LATE-CHR-AFW release category, for which it is known that AFW is already available (adding another AFW pump would have no impact).

Table 3.c-1 below identifies the limitations on the types of events that could be impacted by each SAMA and summarizes their maximum potential risk reductions. Table 3.c-2 below summarizes the bounding net values that were calculated for each SAMA using the 95th percentile PRA results. None of the SAMAs were cost-effective.

Table 3.c-1: Estimated Bounding Impact of SAMAs

SAMA	Limitations	Potential Bounding Risk Reduction Estimate
2: Replace one Low Pressure Condensate Pump with a High Pressure Motor Driven Pump (or Add a Booster Pump) and Add Hotwell Makeup Controls to the MCR from a Non-CST Source	Not available for LOOP, which comprises 30.2% of CDF and 35.8% of the "composite" Level 2 frequency. Not available for "LOOP post-trip", which accounts for 7.1% of CDF and 9.0% of the 'composite' Level 2 frequency. Would not mitigate Loss of all Condensate Pumps: 7.5% of CDF and 8.2% of the "composite" Level 2 frequency. Would not mitigate ATWS with unfavorable moderator coefficient: 7.3% of CDF. Would not mitigate SLOCA with loss of primary side injection: >= 3.3% of CDF and 3.0% of the 'composite' Level 2 frequency. Would not mitigate MLOCA: 2.5% of CDF. Would not mitigate Loss of Turbine Cooling Water events: 2.1% of CDF and 2.2% of the 'composite' Level 2 frequency. Would not mitigate LLOCA: 0.4% of CDF. Would not mitigate ISLOCA: 0.3% of CDF. Other smaller contributors	<39.3% of CDF <41.8% of composite Level 2 frequency
3: Install an Independent AFW System with a Dedicated Power Supply	ATWS with unfavorable moderator coefficient, MLOCA, and LLCOA alone comprise over 10% of CDF, which this AFW system cannot mitigate. For Level 2, about 30 percent of dose-risk is represented by the "LATE-CHR-AFW" release category, for which AFW is already available.	< 90% of CDF, all release category frequencies set to 0.0 apart from LATE-CHR-AFW (retains 3.96 person-rem and $128)

Table 3.c-1: Estimated Bounding Impact of SAMAs

SAMA	Limitations	Potential Bounding Risk Reduction Estimate
7: Add Auto-Start Capability to AFN-P01 on Low SG Level and an Automatic Power Transfer Switch to Address Loss of MFW Cases with Div 1 Power Failures and Operator Start Errors	Not available for SBO, which comprises 23.3% of CDF and 30.5% of the "composite" Level 2 frequency. Would not mitigate ATWS with unfavorable moderator coefficient: 7.3% of CDF. Would not mitigate CCF of electric AFW pumps to start: 6.1% of CDF and 7.2% of the "composite" Level 2 frequency. Would not mitigate MLOCA: 2.5% of CDF Would not mitigate CCF of electric AFW pumps to run: 2.5% of CDF and 3.0% of the "composite" Level 2 frequency. Other smaller contributors	<57.2% of CDF <59.3% of composite Level 2 frequency
9: Install a Backup Control Element Assembly Drive Mechanism	Only address ATWS, which is less than 10% of CDF and a negligible change to the composite L2 frequency..	<10% of CDF, negligible change to the composite L2 frequency

Table 3.c-2: Bounding Net Values for SAMAs No Longer Screened in the Phase I Re-Analysis

SAMA ID	Percent Reduction in CDF	Revised CDF	Percent Reduction in L2 Composite Freq.	Revised Dose-Risk	Revised OECR	Bounding Averted Cost-Risk	95th Percentile Averted Cost Risk	Cost of Imp.	Bounding Net Value
2	39.3%	3.08E-06	41.8%	7.93	$8,689	$2,410,155	$6,507,419	$6,600,000	-$92,581
3	90%	5.07E-07	NA	3.96	$128	$4,814,955	$13,000,379	$15,000,000	-$1,999,621
7	57.2%	2.17E-06	59.3%	5.54	$6,076	$3,434,715	$9,273,731	$9,801,762	-$528,031
9	10%	4.56E-06	0.0%	13.62	$14,929	$107,346	$289,834	$14,215,017	-$13,925,183

Phase II Screening Using Detailed Analysis

In order to assess the impact of the increased external events multiplier on the SAMAs requiring Phase II quantifications, the quantification methodology must be revisited. As described in Section D.6 of the ER, after calculating the external events cost-risk using the multiplier of 2, the cost-risk was distributed among the external events contributors by assuming that the fire cost-risk was equal to the product of the internal events cost-risk and the ratio of the fire CDF to the internal events CDF. The balance of the external events risk was assigned to the non-fire external events and a multiplier based on their fractional contribution to the total external events cost-risk was used to account for their contributions in the Phase II evaluations. Now, however, the seismic CDF is larger than the fire CDF and the external events cost-risk is assumed to be distributed among the external events contributors in proportion to their CDFs.

Consequently, the fire cost-risk would be the product of the internal events cost-risk and the ratio of the fire CDF to the internal events CDF. Of course, because of the assumptions made in the ER about the distribution of the external events CDF, the result is the same for the fire contributor (excluding minor differences related to rounding):

$$\text{Cost-Risk}_{\text{Fire}} = \$778{,}000 * 2.72\text{E-}06 / 5.07\text{E-}06 = \$417{,}389$$

Because the underlying Phase II PRA quantifications are not impacted by the change to the external events multiplier, those results have been retained. As in the ER, the fire cost-risk for each SAMA is quantified by multiplying the base case fire cost-risk (now $417,389) by the ratio of the SAMA fire CDF to the base fire CDF.

For the non-fire external events contributors, the multiplier has been updated to be the ratio of the non-fire External Events CDF to the internal events CDF:

$$\text{Non-Fire EE Multiplier} = (7.56\text{E-}06 - 2.72\text{E-}06) / 5.07\text{E-}06 = 0.955$$

For each SAMA, the non-fire external events cost-risk is the product of the internal events cost-risk and the non-fire external events multiplier of 0.955.

For SAMAs 4, 5, 6, 8, 10, 11, 12, 13, 14, 15, and 23, the external events cost-risks have been recalculated using the above methodology in conjunction with the original results presented in the ER. The updated external events cost-risks were then combined with the original internal events cost-risks to obtain "updated total cost-risk" values. The "per unit" values have been scaled up by a factor of 3 to obtain the "site" values. The "updated averted cost-risk" is the difference between the updated MACR of $5,835,000 and the "updated total cost-risk". Finally, the "updated net value" is determined by subtracting the cost of implementation from the "updated averted cost-risk".

For SAMA 17, the ER presents the base and SAMA SGTR release category frequencies and assumes that the fire contribution is reduced in proportion to the ratio of the SGTR frequencies. In this case, the ratio is 2.25E-07 to 2.53E-07 or, 0.889. This corresponds to a fire CDF of 2.42E-06/yr. Otherwise, the SAMA 17 evaluation is similar to the others.

For the fire based SAMAs (19, 20, 21, and 22), the original ER results are not impacted given that 1) the fire contributions were not impacted by the use of the updated external events multiplier (apart from rounding differences), and 2) the other event types do not contribute to the averted cost-risk for these SAMAs. The averted cost-risk values for these SAMAs are taken directly from the ER.

Table 3.c-3 below summarizes these results for the "point estimate" PRA results case and Table 3.c-4 below summarizes the results for the 95th percentile PRA case. The averted cost-risk values for the 95th percentile PRA results case are obtained by multiplying the "point estimate" averted cost-risk values by a factor of 2.7, as described in section D.7.6 of the ER.

The updated net values in Table 3.c-3 indicate that SAMA 6 has a positive net value while the net values for all of the other SAMAs are negative. The ER only identified SAMA 6 as potentially cost beneficial when the 95th percentile PRA results were applied. When the 95th percentile PRA results are combined with the revised external events multipliers, SAMAs 17 and 23 also have positive net values as shown on Table 3.c-4. As stated in APS Response to RAI 5.c and shown in Enclosure 2, APS is committing to implement SAMAs 6, 17, and 23. The use of the revised external events multiplier has not identified any additional potentially cost beneficial SAMAs.

Table 3.c-3: Results Summary for Revised External Events Multiplier (Point Estimate PRA Results)

SAMA ID	Cost of Implement-ation	ER Internal Events Cost-Risk (unit)	Updated Non-Fire Cost-Risk (unit)	ER SAMA Fire CDF	Ratio of SAMA Fire CDF to Base Fire CDF	Updated Fire Cost-Risk (unit)	Updated Total Site Cost-Risk (unit * 3)	Updated Averted Cost-Risk (site)	Updated Net Value (site)
SAMA 4	$5,498,862	$552,127	$527,281	2.67E-06	9.82E-01	$409,876	$4,467,852	$1,367,148	-$4,131,714
SAMA 5	$6,801,762	$636,573	$607,927	1.04E-06	3.82E-01	$159,443	$4,211,829	$1,623,171	-$5,178,591
SAMA 6	$363,374	$697,669	$666,274	2.72E-06	1.00E+00	$417,389	$5,343,996	$491,004	$127,630
SAMA 8	$3,125,000	$608,669	$581,279	2.52E-06	9.26E-01	$386,502	$4,729,350	$1,105,650	-$2,019,350
SAMA 10	$3,000,000	$729,329	$696,509	2.58E-06	9.49E-01	$396,102	$5,465,820	$369,180	-$2,630,820
SAMA 11	$3,000,000	$753,114	$719,224	2.72E-06	1.00E+00	$417,389	$5,669,181	$165,819	-$2,834,181
SAMA 12	$6,801,762	$680,019	$649,418	2.09E-06	7.68E-01	$320,555	$4,949,976	$885,024	-$5,916,738
SAMA 13	$3,000,000	$770,681	$736,000	2.72E-06	1.00E+00	$417,389	$5,772,210	$62,790	-$2,937,210
SAMA 14	$6,647,190	$760,324	$726,109	2.68E-06	9.85E-01	$411,128	$5,692,683	$142,317	-$6,504,873
SAMA 15	$1,642,698	$708,263	$676,391	2.64E-06	9.71E-01	$405,285	$5,369,817	$465,183	-$1,177,515
SAMA 17	$410,473	$763,908	$729,532	2.42E-06	8.89E-01	$371,059	$5,593,497	$241,503	-$168,970
SAMA 19	$4,661,682	NA	NA	NA	NA	NA	NA	$177,645	-$4,484,037
SAMA 20	$3,625,692	NA	NA	NA	NA	NA	NA	$82,569	-$3,543,123
SAMA 21	$3,365,514	NA	NA	NA	NA	NA	NA	$15,012	-$3,350,502
SAMA 22	$3,272,100	NA	NA	NA	NA	NA	NA	$13,761	-$3,258,339
SAMA 23	$415,620	$753,802	$719,881	2.72E-06	1.00E+00	$417,389	$5,673,216	$161,784	-$253,836

Table 3.c-4: Results Summary for Revised External Events Multiplier (95th Percentile PRA Results)

SAMA ID	Cost of Implementation	Updated Averted Cost-Risk (Point Estimate, site)	Updated Averted Cost-Risk (95th percentile, site)	Updated Net Value (95th percentile, site)
SAMA 4	$5,498,862	$1,367,148	$3,691,300	-1,807,562
SAMA 5	$6,801,762	$1,623,171	$4,382,562	-2,419,200
SAMA 6	$363,374	$491,004	$1,325,711	962,337
SAMA 8	$3,125,000	$1,105,650	$2,985,255	-139,745
SAMA 10	$3,000,000	$369,180	$996,786	-2,003,214
SAMA 11	$3,000,000	$165,819	$447,711	-2,552,289
SAMA 12	$6,801,762	$885,024	$2,389,565	-4,412,197
SAMA 13	$3,000,000	$62,790	$169,533	-2,830,467
SAMA 14	$6,647,190	$142,317	$384,256	-6,262,934
SAMA 15	$1,642,698	$465,183	$1,255,994	-386,704
SAMA 17	$410,473	$241,503	$652,058	241,585
SAMA 19	$4,661,682	$177,645	$479,642	-4,182,041
SAMA 20	$3,625,692	$82,569	$222,936	-3,402,756
SAMA 21	$3,365,514	$15,012	$40,532	-3,324,982
SAMA 22	$3,272,100	$13,761	$37,155	-3,234,945
SAMA 23	$415,620	$161,784	$436,817	21,197

NRC RAI 4.a

Provide the following information concerning the MELCOR Accident Consequence System 2 analyses:

Section D.2.2 states, "A Level 3 (Dose Consequence) analysis was done to support the IPE, but has not been maintained." Section D.3 describes the Level 3 analysis performed for the SAMA evaluation. Clarify the relationship between the two Level 3 analyses. In the response, specifically address whether the Level 3 analysis performed for the SAMA evaluation is a completely new analysis or an update to the IPE analysis.

APS Response to RAI 4.a

The statement made in section D.2.2 about the 1991 IPE Level 3 was intended to convey the need for a new PRA Level 3 model. A new Level 3 model was performed and was based on: (1) MAACS2 Code Sample problem A; (2) March 2007 cost of evacuation; (3) year 2000 census data with growth projections for year 2040; and (4) year 2002 national census of agriculture. The NUREG-1150 1989 food model was used in the new analysis. Non-farm land property values were taken from "Effects of Eliminating the Distinction Between Full Cash and Net Limited Property Value on Property Tax," September 12, 2003 (Ref. Arizona 2003 in Section D.11 of the ER).

NRC RAI 4.b

Provide the following information concerning the MELCOR Accident Consequence System 2 analyses:

Section D.3.1, Supplement 1, describes the projected population growth as "using an exponential growth rate." However, the overall growth rates appear to be exponential only from years 1980 to ~2005, and then approaches a more linear growth (and even tapers off) from years 2005 to 2040. Discuss how the population estimates were developed for the various timeframes and clarify what is meant by "exponential growth rate."

APS Response to RAI 4.b

The total population for each of the contributing counties was taken from the U.S. Census Bureau for the year 2000 (Ref. USCB 2000 in ER Section D.11). Projected populations for each county for the year 2040 were taken from the Arizona Department of Economic Security (Ref. Arizona 2006 in ER Section D.11). From these data, the exponential growth rate, r, was calculated using the equation, $P_{2040} = P_{2000} \times e^{(r \times 40)}$. This is the standard equation used for population growth. Although human population growth is sometimes represented by a linear equation, the use of the exponential equation conservatively projects a larger population. The results of that calculation are provided in Table 4.b-1 below.

Table 4.b-1 Projected County Growth Rates (r)

Name	2000	2040	r
La Paz	19,715	29,715	0.010
Maricopa	3,072,149	7,009,664	0.021
Pinal	179,727	1,081,737	0.045
Yavapai	167,517	390,954	0.021
Yuma	160,026	351,299	0.020

These growth rates were then applied to the sector populations developed by SECPOP2000, as incremented by potential transient populations. In the Palo Verde case, there were no significant transients according to the plant's evacuation time estimate (Ref. Maricopa 2005 in ER Section D.11). In cases where a sector consisted of more than one county, an area-fraction weighting factor was applied to the projected county contributions.

NRC RAI 4.c.i

Provide the following information concerning the MELCOR Accident Consequence System 2 analyses:

Section D.3.4 describes the population evacuation assumptions used for the SAMA analysis.

Provide a table of the sector population distribution within the 10 mile emergency planning zone (EPZ), and out to 50 miles at 10 mile intervals, for the projected population in year 2040.

APS Response to RAI 4.c.i

Table 4.c.i-1 below provides the sector population distribution within the 10 mile emergency planning zone (EPZ), and out to 50 miles at 10 mile intervals, for the projected population in year 2040. Table 4.c.1-2 below provides the key for identifying the sectors.

Table 4.c.i-1
Population Projections by Sector

Sector	Number of Counties	2000 Population	2040 Population
1	1	0	0
2	1	0	0
3	1	0	0
4	1	0	0
5	1	0	0

Table 4.c.i-1 (cont.)
Population Projections by Sector

Sector	Number of Counties	2000 Population	2040 Population
6	1	0	0
7	1	0	0
8	1	0	0
9	1	0	0
10	1	0	0
11	1	0	0
12	1	0	0
13	1	0	0
14	1	0	0
15	1	0	0
16	1	0	0
17	1	23	52
18	1	18	41
19	1	2	5
20	1	0	0
21	1	0	0
22	1	0	0
23	1	0	0
24	1	0	0
25	1	0	0
26	1	0	0
27	1	3	7
28	1	0	0
29	1	0	0
30	1	0	0
31	1	0	0
32	1	0	0
33	1	136	310
34	1	25	57
35	1	128	292
36	1	12	27
37	1	13	30
38	1	0	0
39	1	6	14
40	1	0	0
41	1	0	0
42	1	0	0
43	1	0	0
44	1	0	0
45	1	0	0
46	1	0	0
47	1	0	0

Table 4.c.i-1 (cont.)
Population Projections by Sector

Sector	Number of Counties	2000 Population	2040 Population
48	1	44	100
49	1	25	57
50	1	53	121
51	1	0	0
52	1	112	255
53	1	0	0
54	1	31	71
55	1	4	9
56	1	0	0
57	1	0	0
58	1	0	0
59	1	5	11
60	1	0	0
61	1	0	0
62	1	0	0
63	1	0	0
64	1	11	25
65	1	18	41
66	1	103	235
67	1	123	280
68	1	73	166
69	1	15	34
70	1	58	132
71	1	8	18
72	1	0	0
73	1	0	0
74	1	0	0
75	1	0	0
76	1	0	0
77	1	0	0
78	1	0	0
79	1	9	21
80	1	45	103
81	1	172	392
82	1	119	271
83	1	260	593
84	1	416	948
85	1	349	796
86	1	182	415
87	1	321	732
88	1	79	180
89	1	5	11
90	1	4	9

Table 4.c.i-1 (cont.)
Population Projections by Sector

Sector	Number of Counties	2000 Population	2040 Population
91	1	4	9
92	1	16	36
93	1	0	0
94	1	30	68
95	1	70	160
96	1	255	581
97	1	44	100
98	1	0	0
99	1	14	32
100	1	699	1,594
101	1	11,226	25,595
102	1	231	527
103	1	51	116
104	1	74	169
105	1	0	0
106	1	0	0
107	1	0	0
108	1	26	59
109	1	169	385
110	1	241	549
111	1	78	178
112	1	27	62
113	1	78	178
114	1	366	834
115	1	1,048	2,389
116	1	12,209	27,837
117	1	23,076	52,613
118	1	1,212	2,763
119	1	25	57
120	1	84	192
121	1	6	14
122	1	55	125
123	1	13	30
124	2	0	0
125	3	95	203
126	2	9	20
127	1	4	9
128	1	12	27
129	1	2,235	5,096
130	1	2,177	4,964
131	1	22,344	50,944
132	1	290,497	662,333
133	2	123,278	283,135

Table 4.c.i-1 (cont.)
Population Projections by Sector

Sector	Number of Counties	2000 Population	2040 Population
134	2	65	151
135	1	6	14
136	1	2,350	5,358
137	1	212	483
138	1	42	96
139	2	258	584
140	2	18	40
141	2	0	0
142	2	32	49
143	2	97	198
144	1	68	155
145	2	4,981	11,548
146	2	492	1,140
147	2	270	619
148	1	605,945	1,381,555
149	1	458,925	1,046,349
150	2	1,101	5,285
151	2	75	192
152	1	0	0
153	1	0	0
154	1	2	5
155	2	123	273
156	1	122	268
157	2	0	0
158	1	314	474
159	2	629	1,000
160	3	895	2,046

(Ref. TtNUS 2006 in ER Section D.11)

Table 4.c.i-2
Sector IDs

Direction

N	1	17	33	49	65	81	97	113	129	145
NNE	2	18	34	50	66	82	98	114	130	146
NE	3	19	35	51	67	83	99	115	131	147
ENE	4	20	36	52	68	84	100	116	132	148
E	5	21	37	53	69	85	101	117	133	149
ESE	6	22	38	54	70	86	102	118	134	150
SE	7	23	39	55	71	87	103	119	135	151
SSE	8	24	40	56	72	88	104	120	136	152
S	9	25	41	57	73	89	105	121	137	153
SSW	10	26	42	58	74	90	106	122	138	154
SW	11	27	43	59	75	91	107	123	139	155
WSW	12	28	44	60	76	92	108	124	140	156
W	13	29	45	61	77	93	109	125	141	157
WNW	14	30	46	62	78	94	110	126	142	158
NW	15	31	47	63	79	95	111	127	143	159
NNW	16	32	48	64	80	96	112	128	144	160
rad (mi)	*1*	*2*	*3*	*4*	*5*	*10*	*20*	*30*	*40*	*50*

Radius (miles)

NRC RAI 4.c.ii

Provide the following information concerning the MELCOR Accident Consequence System 2 analyses:

Section D.3.4 describes the population evacuation assumptions used for the SAMA analysis.

The scaled evacuation speed for year 2040 is ~13 percent lower than the base evacuation speed. However, the general population growth is roughly double from year 2006 to year 2040. Clarify this discrepancy.

APS Response to RAI 4.c.ii

The 10-mile EPZ population for Palo Verde for 2005 is 6,644 (see population table in APS Response to RAI 4.c.iii), according to the 2005 evacuation time estimate. The 2040 population for the 10-mile EPZ is 7,715, as summed from the data in Table 4.c.i-1. This is a difference of 1,071 and a percent difference of 16%. This compares well with the percent difference between the evacuation speed estimates of 3.4 m/s for 2005 and 2.93 m/s for 2040, resulting in 16% difference, when comparing to the lower value as was done for the population (necessary because the relationship between population and evacuation speed is inverse).

NRC RAI 4.c.iii

Provide the following information concerning the MELCOR Accident Consequence System 2 analyses:

Section D.3.4 describes the population evacuation assumptions used for the SAMA analysis.

Provide the reference year EPZ population for the evacuation study (2005).

APS Response to RAI 4.c.iii

The 10-mile EPZ population from the evacuation study is shown in Table 4.c.iii-1 below (Ref. Maricopa 2005 in ER Section D.11).

Table 4.c.iii-1 2005 Population Survey

Sector	1 Mile	2 Mile	2 Mile TOTAL	3 Mile	4 Mile	5 Mile	5 Mile TOTAL	6 Mile	7 Mile	8 Mile	9 Mile	10 Mile	**10** Mile TOTAL
A	0	37	37	123	141	134	435	65	68	116	70	108	862
B	0	15	15	104	224	116	459	101	37	212	87	3	899
C	0	0	0	52	54	266	372	290	86	16	35	18	817
D	0	0	0	14	92	264	370	213	101	5	25	675	1389
E	0	0	0	9	27	114	150	172	24	29	101	82	558
F	0	0	0	3	143	80	226	30	55	25	41	106	483
G	0	0	0	0	5	3	8	3	168	82	12	3	276
H	0	0	0	3	40	0	43	0	0	2	39	26	110
J	0	0	0	30	0	3	33	0	0	0	0	0	33
K	0	0	0	3	0	2	5	2	0	0	0	0	7
L	0	9	9	3	10	17	39	0	2	0	0	2	43
M	6	2	8	0	0	0	8	22	9	0	0	0	39
N	9	0	9	14	0	0	23	0	0	0	0	0	23
P	3	0	3	0	0	0	3	0	0	0	0	0	3
Q	19	5	24	0	3	12	39	15	17	14	45	221	351
R	4	13	17	12	122	88	239	6	14	288	63	141	751
Mile Ring Total	41	81	122	370	861	1099	2452	919	581	789	518	1385	6644

NRC RAI 5.a.i

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

In estimating the benefit of the reduction in risk from external events, PVNGS provides a separate analysis to estimate the benefit of the reduction in fire risk. Since a Level 2 type model was not developed for the fire model, the benefit from the reduction in fire risk is calculated by multiplying the reduction in fire CDF by the maximum internal events benefit (assumes external risk = internal risk). However, this approach is not necessarily conservative for SAMAs in which the benefit is dominated by the reduction in population dose risk or off-site economic cost risk (OECR) and not CDF. This is the case for SAMAs 4, 8, and 15.

Explain why the percent reduction in dose-risk and OECR for SAMAs 4, 8, and 15 are so much greater than the percent change in CDF.

APS Response to RAI 5.a.i

The percent reduction in plant CDF does not necessarily translate into a proportionate reduction in dose risk from all 12 release categories (listed in the response to RAI 2.e.ii). The impact could be a lower reduction in dose risk when the affected release category contributes a small fraction to the total releases. Conversely, the reduction in dose risk could be higher than the corresponding reduction in plant CDF when the affected release category represents a large fraction of all releases.

The disproportionate reduction in dose-risk and OECR relative to the CDF is due to the distribution of the change in CDF among the release categories and their corresponding consequential doses and offsite economic costs. For SAMAs 4, 8, and 15, the CDF reductions correlate to relatively large reductions in the frequencies of the most influential release categories (i.e., "LATE-CHR-NOAFW" and "LATE-CHR-AFW").

For example, in SAMA 4, the total CDF is reduced by 1.03E-6/yr, which is 20.3% of the base CDF of 5.07E-06/yr. Part of this CDF reduction, 1.68E-07/yr, occurs in the "LATE-CHR-NOAFW" release category. This represents only 3.3% of the total CDF (1.68E-07 / 5.07E-06 * 100 = 3.3%); however, it is 31% of the "LATE-CHR-NOAFW" release category frequency (1.68E-07 / 5.46E-07 * 100 = 31%). Because "LATE-CHR-NOAFW" contributes 46% of the total dose-risk, this corresponds to a 14% change in the total dose-risk (0.31 * 0.46 * 100 = 14%). In summary, a 3.3% reduction in CDF correlates to about a 14% reduction in dose-risk.

The behavior of the "LATE-CHR-AFW" release category is similar.

The contributions to dose-risk and OECR from any given release category are, of course, the product of the frequency and conditional consequences for the release category. In order to clarify how these factors are combined for PVNGS, the following table is provided:

PVNGS Release Category Specific Results

Release Category	INTACT	LATE-BMMT-AFW	LATE-BMMT-NOAFW	LATE-BMMT-PDS2	LATE-CHR-AFW	LATE-CHR-NOAFW	LATE-CHR-PDS2	LERF-BYPASS	LERF-ISO	LERF-CFE	LERF-SGTR	Sum of Annual Risk
Conditional Dose	1.34E+02	5.56E+03	2.43E+05	1.13E+05	3.09E+06	1.15E+07	2.47E+06	9.33E+06	1.75E+07	7.14E+06	8.96E+06	**NA**
Conditional OEC	$0.00E+00	$9.27E+01	$4.62E+06	$1.98E+06	$1.00E+08	$1.86E+10	$9.23E+07	$2.96E+10	$2.47E+10	$1.32E+10	$1.56E+10	**NA**
Freq. (per yr)$_{BASE}$	1.720E-07	4.920E-07	1.850E-06	5.010E-07	1.280E-06	5.460E-07	1.210E-07	1.510E-08	9.330E-09	0.000E+00	2.530E-07	**5.24E-06**
Dose-Risk$_{BASE}$	0.00	0.00	0.45	0.06	3.96	6.28	0.30	0.14	0.16	0.00	2.27	**13.62**
OECR$_{BASE}$	$0	$0	$9	$1	$128	$10,156	$11	$447	$230	$0	$3,947	**$14,929**

NRC RAI 5.a.ii

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

In estimating the benefit of the reduction in risk from external events, PVNGS provides a separate analysis to estimate the benefit of the reduction in fire risk. Since a Level 2 type model was not developed for the fire model, the benefit from the reduction in fire risk is calculated by multiplying the reduction in fire CDF by the maximum internal events benefit (assumes external risk = internal risk). However, this approach is not necessarily conservative for SAMAs in which the benefit is dominated by the reduction in population dose risk or off-site economic cost risk (OECR) and not CDF. This is the case for SAMAs 4, 8, and 15.

Provide revised cost-benefit evaluations for SAMAs 4, 8, and 15 that account for the higher reduction in dose-risk and OECR than CDF.

APS Response to RAI 5.a.ii

As described in the APS Response to RAI 5.a.i, the disproportionate reduction in dose-risk and OECR relative to the CDF is due to the distribution of the change in CDF among the release categories and their corresponding consequential doses and offsite economic costs. The fire model is not integrated with the current Level 2 model and similar insights related to the Level 3 impacts of a SAMA are not available. Because these insights are not available, the ER assumed that the fire cost-risk was equal to the ratio of the fire CDF to the internal events CDF multiplied by the internal events cost-risk. The implicit assumption is that the fire CDF is distributed among the internal events release categories in the same proportion as the internal events CDF. It is known that this is not necessarily the case, but it is used as an approximation.

Deriving the averted cost-risk for a fire SAMA based on the change in CDF, as performed in the ER, is limited in that it does not account for the eventuality that a SAMA could result in a change in CDF that maps to some of the more influential release categories rather than more uniformly across all of the release categories. Use of the "non-fire" external events multiplier to address the other external events contributors, however, accounts for the increased impact on the Level 3 results. This is because the multiplier is applied to an averted cost-risk that is based on both the change in CDF as well as the Level 3 changes.

The large changes in dose-risk and OECR relative to the CDF changes for SAMAs 4, 8, and 15 is an indication that the CDF changes map to the more influential release categories. In order to account for the increased impact on the Level 3 results for these SAMAs, the fire averted cost-risk will be re-calculated using information from the internal events Level 3 results. Specifically, the fire cost-risk for each SAMA will be calculated by multiplying the baseline fire cost-risk by the smaller of the following two ratios:

$$\text{Dose-Risk}_{SAMA} / \text{Dose-Risk}_{Base} \quad \text{or} \quad \text{OECR}_{SAMA} / \text{OECR}_{Base}$$

The fire averted cost-risk and the SAMA's net value will then be calculated in the same manner as described in the ER. This is performed for each of the three SAMAs identified below. As documented in the APS Response to RAI 3.c, the use of the revised external events multiplier has no impact on the fire cost-risk beyond rounding differences. Consequently, the results from the ER are used as the basis for these re-quantifications.

The non-fire external events contributions and the MACR, however, are impacted by the revised external events multiplier and the results from the APS Response to RAI 3.c are used in place of the results documented in the ER.

SAMA 4: SBO Mitigation (SBO Gas Turbine Generators [GTGs] not available)

For SAMA 4, the dose-risk ratio is the smaller of the two ratios:

SAMA 4 Level 3 Ratio Results

	Dose-Risk	**OECR**
Base Results	13.62	$14,929
SAMA Results	8.94	$11,204
Ratio $_{SAMA/Base}$	0.656	0.750

Consequently, the revised fire cost-risk is $273,557 ($417,008 * 0.656 = $273,557). Using the results from Section D.6.1.1 of the ER and the response to RAI 3.c, the total SAMA 4 cost risk can be re-calculated:

Revised SAMA 4 Total Cost-Risk

Internal Events Cost-Risk	Non-Fire External Events Cost-Risk	Fire Cost-Risk	Multiplier for Three Units	Total Cost-Risk (Site, SAMA Implemented)
$552,127	$527,281	$273,557	3	$4,058,895

The averted cost-risk is the difference between the total base case cost-risk and the total cost-risk (provided on a site basis):

Revised SAMA 4 Averted Cost-Risk

Base Case Total Cost-Risk (MACR)	Total SAMA Cost-Risk	Averted Cost-Risk
$5,835,000	$4,058,895	$1,776,105

The net value for this SAMA is the difference between the averted cost-risk and the cost of implementation:

Revised SAMA 4 Net Value

Averted Cost-Risk	Cost of Implementation	Net Value
$1,776,105	$5,498,862	-$3,722,757

Even with the changes to account for the increased impact on the Level 3 results, SAMA 4 still has a large negative net value. If the 95th percentile PRA results are applied, the averted cost-risk of $1,776,105 would be increased by a factor of 2.7 to a value of $4,795,484. The net value is still negative at -$703,378 ($4,795,484 - $5,498,862 = -$703,378).

SAMA 8: Add Auto start/Load Capability to the GTGs

For SAMA 8, the OECR ratio is the smaller of the two ratios:

SAMA 8 Level 3 Ratio Results

	Dose-Risk	OECR
Base Results	13.62	$14,929
SAMA Results	10.60	$10,442
Ratio $_{SAMA/Base}$	0.778	0.699

Consequently, the revised fire cost-risk is $291,489 ($417,008 * 0.699 = $291,489). Using the results from Section D.6.3.1 of the ER, the total SAMA 8 cost risk can be re-calculated:

Revised SAMA 8 Total Cost-Risk

Internal Events Cost-Risk	Non-Fire External Events Cost-Risk	Fire Cost-Risk	Multiplier for Three Units	Total Cost-Risk (Site, SAMA Implemented)
$608,669	$581,279	$291,489	3	$4,444,311

The averted cost-risk is the difference between the total base case cost-risk and the total cost-risk (provided on a site basis):

Revised SAMA 8 Averted Cost-Risk

Base Case Total Cost-Risk (MACR)	Total SAMA Cost-Risk	Averted Cost-Risk
$5,835,000	$4,444,311	$1,390,689

The net value for this SAMA is the difference between the averted cost-risk and the cost of implementation:

Revised SAMA 8 Net Value

Averted Cost-Risk	Cost of Implementation	Net Value
$1,390,689	$3,125,000	-$1,734,311

Even with the changes to account for the increased impact on the Level 3 results, SAMA 8 still has a large negative net value. If the 95th percentile PRA results are applied, the averted cost-risk of $1,390,689 would be increased by a factor of 2.7 to a value of $3,754,860. The net value is positive at $629,860 ($3,754,860 - $3,125,000 = $629,860).

SAMA 15: 100% Capacity Battery Chargers

For SAMA 15, the OECR ratio is the smaller of the two ratios:

SAMA 15 Level 3 Ratio Results

	Dose-Risk	OECR
Base Results	13.62	$14,929
SAMA Results	12.36	$13,150
Ratio $_{SAMA/Base}$	0.907	0.881

Consequently, the revised fire cost-risk is $367,384 ($417,008 * 0.881 = $367,384). Using the results from Section D.6.7.1 of the ER, the total SAMA 8 cost risk can be re-calculated:

Revised SAMA 15 Total Cost-Risk

Internal Events Cost-Risk	Non-Fire External Events Cost-Risk	Fire Cost-Risk	Multiplier for Three Units	Total Cost-Risk (Site, SAMA Implemented)
$708,263	$676,391	$367,384	3	$5,256,114

The averted cost-risk is the difference between the total base case cost-risk and the total cost-risk (provided on a site basis):

Revised SAMA 15 Averted Cost-Risk

Base Case Total Cost-Risk (MACR)	Total SAMA Cost-Risk	Averted Cost-Risk
$5,835,000	$5,256,114	$578,886

The net value for this SAMA is the difference between the averted cost-risk and the cost of implementation:

Revised SAMA 15 Net Value

Averted Cost-Risk	Cost of Implementation	Net Value
$578,886	$1,642,698	-$1,063,812

Even with the changes to account for the increased impact on the Level 3 results, SAMA 15 still has a large negative net value. If the 95th percentile PRA results are applied, the averted cost-risk of $578,886 would be increased by a factor of 2.7 to a value of $1,562,992. The net value is still negative at -$79,706 ($1,562,992 - $1,642,698 = -$79,706).

Summary

Calculating the fire averted cost-risk using CDF information alone can underestimate the averted cost-risk for a SAMA when its implementation impacts the more important release categories. When the larger impact on the Level 3 results is accounted for in conjunction with the revised external events multiplier, the net values for SAMAs 4, and 15 remain negative even when the 95th percentile PRA results are considered. SAMA 8, however, becomes potentially cost effective when the 95th percentile PRA results are combined with the revised external events multiplier.

NRC RAI 5.b

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

SAMAs 6, 11, and 13 have no fire CDF reduction. Explain why these SAMAs (the spurious bus lockout events in SAMA 6, loss of essential cooling water flow to the shutdown cooling heat exchangers in SAMA 11, and loss of turbine-building cooling water in SAMA 13) do not impact internal fire CDF.

APS Response to RAI 5.b

The two events addressed in SAMAs 6 and 13 have very low contribution to the PVNGS Fire CDF. Neither spurious ESF bus lockout, nor loss of turbine cooling water is among the Fire PRA model initiators. Also, the loss of turbine cooling water is a low contributor to plant internal events CDF (about 0.1%).

It is standard PRA practice to exclude dual initiating events from analysis due to low contribution. Because SAMAs 6 and 13 are tailored to address the risk of specific internal events initiating events, their benefits will never be realized in conjunction with other initiating events, such as internal fires. Specifically, SAMA 6 addresses the loss of an engineered safety features (ESF) bus while SAMA 13 addresses the loss of turbine building cooling water.

For SAMA 11, the intent of the enhancement is to use fire water to provide an alternate means of cooling the shutdown cooling system (SDC) heat exchangers; however, for fire events, credit cannot be taken for using the fire water system for applications other than fire suppression. Consequently, there is no reduction in the fire CDF for SAMA 11.

NRC RAI 5.c

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

The cost of implementation of new or modified procedures for SAMAs 6, 17, and 23 is estimated to be $363,374, $410,473, and $415,620, respectively. These costs are significantly higher than the $50,000 generally used in SAMA analyses and appear to be based on a detailed cost analysis (based on the number of significant figures reported). Section D.5.1.1 states that the reason for this difference is that the scope accounted for in the PVNGS estimates "is greater than the scope corresponding to the types of changes used to establish the minimum expected cost of implementation." Clarify this statement and provide additional justification for these PVNGS estimates.

APS Response to RAI 5.c

APS has decided to implement SAMAs 6, 17, and 23. The commitment to implement these SAMAs is provided in LRA Amendment No. 2 in Enclosure 2. Since these SAMAs will be implemented, there is no need for further cost-benefit evaluation.

NRC RAI 5.d

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

The cost of implementation of SAMAs 5 and 12, install an automatic transfer switch, are assumed to be the same at $2,267,254 per unit. Clarify why the cost of implementation of the non-safety automatic transfer switch in SAMA 5 is the same as for implementation of the safety-related automatic transfer switch in SAMA 12. Justify the cost estimate for these SAMAs.

APS Response to RAI 5.d

The conceptual cost estimates were the same for SAMAs 5 and 12 because, although one system is safety-related and the other is non-safety, the equipment, design and installation are reasonably similar, which is standard practice for estimating at an "order of magnitude" level. If further refinements were made for the cost estimate of the non-safety related location, is it unlikely to result in a substantial reduction. Since the total averted cost risk of SAMA 5 was calculated as $3,763,365, and the implementation cost calculated as $6,801,762, a more in-depth cost analysis of SAMA 5 would not be expected to result in reducing the implementation cost by the 45% that would be needed to result in a positive net value for the SAMA.

The estimate for the installation of automatic transfer switches is a conceptual estimate. It assumed that the design and engineering, as well as the development of design modification work orders, were to be performed by in-house personnel. The estimate also assumed that the actual implementation (field work) was to be performed by contract labor consisting of a foreman and various supporting disciplines (electrical, carpenters and mechanical). Appropriate load rates were applied to labor.

Material prices such as the purchase of transfer switches and associated control panels, and costs of shipment to Palo Verde, were derived from conversation with vendors who would normally supply this type of equipment used in a nuclear power plant application.

Due to the "order of magnitude" level of this estimate, a 20% contingency was added to the whole scope of work. The estimate totaled $2,267,254.00

NRC RAI 5.e

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

SAMA 4 (station blackout mitigation, gas turbine generators not available) has an estimated cost of $1.8M for implementation of a portable 480 V AC generator to power the division 1 station batteries. The implementation costs of similar SAMAs in three other plants were $230,000, $489,000, $494,000. Justify the cost estimate for SAMA 4.

APS Response to RAI 5.e

The scope of PVNGS SAMA 4 is different from the scope of the similar SAMAs at the other three other plants discussed in RAI 5.e. The PVNGS SAMA requires the portable generator to support more equipment than the generators for the similar SAMAs at the other plants.

For PVNGS, the generator must be capable of supporting a battery charger for long term auxiliary feedwater operation in addition to at least two charging pumps for reactor coolant system makeup. The primary side makeup pumps are considered to be required to maintain the plant in a stable state up to, and beyond, the 24 hour mission time. Addressing primary side makeup requires additional design engineering work that is not required for supporting the battery chargers alone. The cost of the generator is also impacted by the increased load that it must carry; each generator is estimated to cost over $640,000 by itself.

The sizes of the portable generators for the similar SAMAs at the other plants would be smaller than those for PVNGS SAMA 4 because they would only be required to carry the load associated with maintaining control power and level instrumentation, and not to provide the motive power for any makeup pumps.

In summary, the difference in the cost of the PVNGS SAMA is justified by the scope of the SAMA, which is larger than any of those cited in RAI 5.e.

NRC RAI 6.a

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER, would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at PVNGS:

Modify procedures to shed component cooling water (CCW) loads on loss of essential raw cooling water to extend component cooling water heat-up time.

APS Response to RAI 6.a

The SAMAs developed for PVNGS were designed to effectively address the plant specific risks in the most cost effective manner possible. No lower cost alternatives have been identified that could provide comparable benefits.

For some plants, loss of the essential raw cooling water system (or its equivalent) will result in failure of both reactor coolant pump (RCP) thermal barrier cooling (seal cooling) and RCP seal injection. RCP seal cooling, which is directly dependent on CCW, is lost due to CCW's dependence on the essential raw water system for heat removal. Seal injection is lost because the charging pumps depend on CCW for pump or lube oil cooling. The PVNGS charging pumps do not require cooling from another system. Consequently, loss of nuclear cooling water, which provides RCP seal cooling, does not impact the charging pumps and seal injection would still be available when nuclear cooling water is lost. In addition, the essential cooling water system is available to provide backup cooling to the nuclear cooling water system. For PVNGS, the loss of nuclear service water initiating event has risk reduction worth (RRW) values of 1.002 and 1.001 for CDF and Level 2, respectively. These are well below the RRW review threshold of 1.01, are not important contributors, and SAMAs related to increasing the reliability or availability of the nuclear service water system require no further evaluation for PVNGS.

NRC RAI 6.b

For certain SAMAs considered in the ER, there may be lower-cost alternatives that could achieve much of the risk reduction at a lower cost. In this regard, discuss whether any lower-cost alternatives to those Phase II SAMAs considered in the ER, would be viable and potentially cost-beneficial. Evaluate the following SAMAs (previously found to be potentially cost-beneficial at other plants), or indicate if the particular SAMA has already been considered. If the latter, indicate whether the SAMA has been implemented or has been determined to not be cost-beneficial at PVNGS:

Install backwash filters in place of existing service water pump discharge strainers to reduce probability of common cause failures.

APS Response to RAI 6.b

The ultimate heat sink for PVNGS is the essential spray pond, which appears to be the approximate equivalent to the service water system at other the plants with the SAMA cited in RAI 6.b. Common cause plugging or blocking of the discharge path in the essential spray pond system at PVNGS has not been identified as a contributor to PVNGS risk. As a result, no SAMAs addressing discharge path plugging would be cost-beneficial.

ENCLOSURE 2

Palo Verde Nuclear Generating Station

License Renewal Application

Amendment No. 2

Source: RAI 3.a.iv

LRA Section D.5.1.6.1, Internal Fires, Table on Page D-39, is revised as follows (new text underlined):

Sequences:	No significant sources of conservatism were identified for PVNGS. Sequences in the PVNGS fire model are defined in detail. The consequences of any sequence collapsing is likely minor.
Level of Detail:	No significant sources of conservatism were identified for PVNGS. Many fire PRAs may have reduced level of detail in the mitigation of the initiating event and consequential system damage; however, the PVNGS model includes a detailed assessment of the impacts of the initiating events, consequential fire damage, and the subsequent response of the plant.

Source: RAI 5.c

LRA Table A4-1, License Renewal Commitments, item no. 49, Page A-58, is revised as follows (new text underlined):

49	APS commits to ~~will consider~~ implement ~~the three~~ SAMAs (6, 17 and 23) ~~identified in the analysis using the appropriate PVNGS design process~~ prior to the period of extended operation. (RCTSAI 3246952)	Environmental Report D.8	~~12/31/09~~ Prior to the period of extended operation[1]

LRA ER Section D.8, Conclusions, Page D-120, is revised as follows (new text underlined):

In summary, three relatively low cost SAMAs (SAMAs 6, 17 and 23) have been identified as cost beneficial and are suggested for ~~potential~~ implementation at PVNGS. ~~While these results are believed to accurately reflect potential areas for improvement at the plant, APS notes that this analysis should not necessarily be considered a formal disposition of these proposed changes as other engineering reviews are necessary to determine the ultimate resolution. APS will consider the three SAMAs (6,17 and 23) identified in the analysis using the appropriate PVNGS design process.~~ APS commits to implement SAMAs 6, 17 and 23 prior to the period of extended operation.