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STN-50-529 Palo Verde Nuclear Station, Unit 2, Arizona Publi 05000529
STN-50-530 Palo Verde Nuclear Station, Unit 3, Arizona Publi 05000530
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SUBJECT: Forwards responses to 970714 RAI re charging sys commitments for PVNGS.

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102-04012- JML/AKK/SAB/GAM
September 16, 1997

U. S. Nuclear Regulatory Commission
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Dear Sirs:

Subject: Palo Verde Nuclear Generating Station (PVNGS)
Units 1, 2, and 3
Docket Nos. STN 50-528/529/530
Responses to July 14, 1997 NRC Request for Additional Information Regarding Charging System Commitments for the Palo Verde Nuclear Generating Station

Enclosed are responses to the NRC request for additional information regarding charging system commitments for the Palo Verde Nuclear Generating Station dated July 14, 1997.

Should you have any questions, please contact Scott A. Bauer at (602) 393-5978.

Sincerely,

Gregg R. Overbeck
for JML.

JML/AKK/SAB/GAM

Enclosure

cc: E. W. Merschoff
K. E. Perkins
K. M. Thomas
NRC Sr. Resident

A001
1/1

Enclosure

Responses to July 14, 1997
NRC Request for Additional Information
Regarding Charging System Commitments
for the Palo Verde Nuclear Generating Station

Responses to July 14, 1997 NRC Request for Additional Information Regarding Charging System Commitments for the Palo Verde Nuclear Generating Station

NRC Question 1:

Please explain any major differences between your current PRA and the PRA used to support your IPE. Among these differences, specifically identify differences related to loss of offsite power (LOOP) sequences and explain the differences and impact of LOOP risk.

PVNGS Response 1:

The engineering study 13-NS-B35 (Appendix A of letter no. 102-03278-WLS/AKK/GAM, March 9, 1995, from APS to NRC) is based on the results of the 1992 PRA model (IPE submittal dated April 28, 1992). The 1994 PRA model (current) had one major change and several minor changes. The major change is the addition of the gas turbine generators (GTGs) to the model. This has a direct impact on the importance of loss of off-site power. PVNGS now has two GTGs for providing alternate power during a station blackout condition. PVNGS also has accumulated a larger base of equipment and initiator history. This allows some of the equipment and initiators modeled in the PRA to be Bayesianed with PVNGS history. In comparing the two models the following statistics were produced:

> LOOP and station blackout (SBO) sequences were reduced by 64% because of the GTGs and human recovery of the auxiliary feedwater N (non-Seismic Category I) train.
>
> Miscellaneous Reactor Trip and Turbine Trip sequences are reduced by 62% and 82%, respectively, due to Bayesian updated data.

NRC Question 2:

Describe any review of the PRA that has been made to ensure that the current PRA represents the as-built, as-operated plant. Discuss any changes made to the PRA due to such reviews, specifically with changes associated with the LOOP contributions.

PVNGS Response 2:

For the IPE (1992 model), the PRA engineers walked down the systems and an internal peer review was performed. For the 1994 quantification, the PRA engineers reviewed design change packages (DCPs) to identify any impact on the PRA due to plant

changes since the 1992 model. Major changes to the model have been in the areas of plant data and new site modifications. The changes to the PRA are described in the PVNGS Response 1 above.

NRC Question 3:

Provide the results of the Fussell-Vesely and Risk Achievement Worth Importance analyses for the APSS/Charging System.

PVNGS Response 3:

Based upon the guidelines used for the Maintenance Rule, the charging (CH) system is ranked high. Table 1 below identifies the events which make up the CH system and which survived a 10E-9 truncation limit. The boric acid makeup (BAM) pumps and the supply line valves were truncated from the 1994 results. As a sensitivity analysis, the event (1BAM-CHGSUC--2OP) was set to true and the 1994 model was re-solved using the Integrated Reliability and Risk Analysis System (IRRAS). There was little or no change to the importance of the system with the BAM pumps set to true.

Table 1: Charging System Events Which Survived a 10E-9 Truncation Limit

	1992 Model Results (SETS)		1994 Model Results (IRRAS)		1994 Model Results (IRRAS)*	
Event	FusVes	AchW	FusVes	Risk Increase	FusVes	Risk Increase
1BAM-CHGSUC--2OP	8.50E-06	1				
1BAM-VCTLINE-2OP	8.50E-06	1			1.26E-04	1.01E+00
1CHAF20---FX-PG	5.25E-04	2.09	1.06E-03	3.20E+00	1.06E-03	3.20E+00
1CHAHV0531-MV-RO	2.85E-04	2.13	3.39E-04	2.34E+00	3.39E-04	2.34E+00
1CHAV306---CV-FC			3.43E-04	1.10E+00	3.43E-04	1.10E+00
1CHBF20---FX-PG	6.17E-04	2.28	9.84E-04	3.05E+00	9.84E-04	3.05E+00
1CHBHV0530-MV-RO	3.27E-04	2.3	3.39E-04	2.34E+00	3.39E-04	2.34E+00
1CHBV305---CV-FC			2.50E-04	1.08E+00	2.50E-04	1.08E+00
1CHLT-226-227-CC	6.99E-06	1.16				
1CH-RWTTEMP--2OP	3.54E-04	1.01	1.45E-02	1.47E+00	1.45E-02	1.47E+00
1CH-SEALINJ--2CM	1.13E-04	1.01	4.53E-03	1.30E+00	4.53E-03	1.30E+00
1CH-SEALINJ--2OP	8.27E-05	1.01	4.58E-04	1.10E+00	4.58E-04	1.10E+00
1PZRSPRAYLIN-2OP	1.53E-04	1.21	8.83E-05	1.12E+00	8.82E-05	1.12E+00
1RWT-CHGLINE-2OP	9.02E-04	1.02	2.38E-02	1.48E+00	2.39E-02	1.49E+00

* - The IRRAS run is based upon setting 1BAM-CHGSUC--2OP to true
Note - A blank field indicates that the event was truncated in that particular analysis.
Note - All quantifications were solved at 1E-9

NRC Question 4:

Appendix A, page 7, of your submittal states that the probability of nonrecovery of offsite power was estimated at 6.0E-3 and 5.0E-4 at 10 hours and 36 hours respectively, and that these values were obtained from Figure A.14 of NUREG-1032. Please explain the applicability of Figure A.14 (estimated frequency of losses of offsite power exceeding specified durations for Limerick) in modeling nonrecovery at your site. Also, since NUREG-1032 was published in 1988, how is the value of CDF given in Appendix A of your submittal (1.7E-7 core damage events/yr) affected by using more recent data on LOOP recovery?

PVNGS Response 4:

Appendix A of the March 9, 1995 submittal references NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issue A-44, Draft Report for Comment," May 1985. Figure A.14 of the May 1985 version of NUREG-1032, entitled "Estimated frequency of occurrence of losses of offsite power exceeding specified duration for nine offsite power clusters," was used in the study that was the basis of Appendix A of the submittal. The study identifies that PVNGS would be in cluster 2 (GR = 2, I = 2, SR = 4, and SS =2).

PVNGS documented in study 13-NS-C04, "Statistical Evaluation of Loss of Off-Site Power Frequency and Duration", new "failure to recover off-site power" numbers based upon NSAC-203 data. In the study, the value for failure to restore after 12 hours was estimated to be .02. As a result of this analysis it would seem that NUREG-1032 values were somewhat optimistic. This new analysis also estimates a new initiator frequency for LOOP. Based upon these new numbers (initiator =.041/yr, 10 hour unrecovery =.02 and 36 hour unrecovery =.01) the CDF estimated for the scenario identified in 13-NS-B35 would be approximately 3.3 times greater.

NRC Question 5:

Please explain how the LOOP initiating event frequency used in the scoping study of page 6 of 7 of Appendix A is calculated. This value is also used in the IPE. Had the initiating event frequency been updated to incorporate additional data attained on LOOP initiating event frequencies?

PVNGS Response 5:

The analysis in the study and the 1992 IPE for the initiator LOOP are the same. The initiator is based upon NSAC-111 data and a point estimate was calculated. For the LOOP initiator used in Response 4, a Two-Stage Population Variability Method was used.

NRC Question 6:

In the scoping study on page 6 of 7 of Appendix A, the analysis only includes depletion of the condensate storage tank (CST) as a failure mechanism for failure of auxiliary feedwater system (AFW) to perform decay-heat removal. Please explain why other failure mechanisms of the AFW system were not included in the analysis.

PVNGS Response 6:

The loss of offsite power (LOOP) event tree, as reported in the IPE and in the current PRA model, does consider auxiliary feedwater (AFW) failure due to other failure mechanisms (i.e., AFW pump fails to start, run, in corrective maintenance). Failure of AFW due to these failure modes leads directly to core melt (CM) in the event tree. Since the previous branch in the LOOP event tree assumes that on a loss of AFW (due to failure mechanisms other than CST depletion) the plant will go to core melt, the event tree presented in Appendix A is only applicable for the AFW success branch. At the point of CST depletion, auxiliary pressurizer spray system (APSS) availability, and off-site power recovery is considered.

NRC Question 7:

In your IPE submittal, dated April 28, 1992, you state that the chemical and volume control system (CVCS)/APSS success criteria for steam generator tube rupture (SGTR) conditions is "at least one charging pump is supplying borated water from the RWT through one of the APSS valves for 8 hours." However, page 3 of the Engineering Evaluation of your March 9, 1995, submittal states that Supplement 9 to the PVNGS SER concluded that the APSS was not needed for mitigation of a design basis SGTR. How is this consistent with your IPE SGTR success criteria? Also, the scoping study on pages 6 and 7 of Appendix A only includes the change in CDF from LOOP initiating event. Please explain how SGTR contributes to a decrease in CDF resulting from an increase in APSS reliability.

PVNGS Response 7:

The success criteria identified in the IPE submittal identifies several alternative methods of supporting a particular safety function. These success paths came from either the EOPs or independent engineering analyses. It was not intended to differentiate between plant design basis and the actual plant capabilities (best estimate). Another example of this type of treatment is the inclusion of the condensate pumps as an alternative to supplying water to the steam generators for several initiators. The result of failing the boric acid makeup (BAM) pumps and associated valves (by setting 1BAM-CHGSUC--2OP to true; see response to question 3) increased the core damage frequency by 7E-9/yr. This increase was seen only in the anticipated transient without scram (ATWS) scenario and not the SGTR sequences. Therefore, improving the

reliability of the APSS/Charging system (by the BAM pump modification), does not decrease CDF with regard to the SGTR. The reason for this is because there are three methods of supporting the safety function of controlling RCS pressure. They are auxiliary pressurizer spray, normal pressurizer spray, and the pressurizer vents. This three element cutset probability is very low, resulting in an insensitive change in SGTR CDF contribution.

NRC Question 8:

On page 5-54 of your IPE, you state that the CVCS success criteria for ATWS conditions is one charging pump supplying 40 gpm of borated water for one hour. The CVCS system includes both charging system and APSS. The scoping study on pages 6 and 7 of Appendix A only includes the change in CDF from the LOOP initiating event. Explain how improved APSS reliability contributes to a decrease in CDF resulting from ATWS.

PVNGS Response 8:

The result of setting 1BAM-CHGSUC--2OP (the boric acid makeup [BAM] pumps and associated valves; see importance analysis, response to question 3) to true increased the core damage frequency by 7E-9/yr. This increase was seen only in the ATWS scenario. The reason for the slight increase in the ATWS scenarios is because charging is an alternate method of getting boron into the core and the BAM pumps are one of two success paths for this safety function.

NRC Question 9:

Also on page 5-54 of your IPE, you state that the success criteria to maintain reactor coolant system (RCS) integrity is seal injection or nuclear cooling water. It is further stated that if loss of nuclear cooling water occurs, the operator must secure the reactor coolant pumps (RCPs) within 10 minutes if seal injection is available or within 5 minutes if seal injection flow is not available. The success criteria for providing seal injection is one charging pump for 24 hours. Since the scoping study in Appendix A of your submittal only includes LOOP initiating event, please explain whether an increase in APSS reliability will decrease the contribution of loss of nuclear cooling water to CDF. If there is an impact, please provide the quantitative CDF estimate for projected changes in APSS reliability given loss of nuclear cooling water.

PVNGS Response 9:

To calculate the impact on the loss of nuclear cooling water (NCW), the power sources to the boric acid makeup (BAM) pumps were replaced with class power (PHAM35), CDF increased by 2E-9/yr. This slight increase is due to dependent power failure modes between the BAM pumps and the charging pumps (Class 4.16kv power losses).

There is also a lesser increase (1E-9/yr) in CDF from ATWS for the same reason. From this same model, however, it should be noted that there is a 1.3E-6/yr decrease in the LOOP RCP seal LOCA sequences. If the process identified in 13-NS-B35 is applied using the new numbers (CDF decrease of 1.3E-6/yr for LOOP and 3.3E-7 for off-site power recovery) the total benefit of the proposed change becomes approximately $90,000.

NRC Question 10:

In your PRA, what is the success criteria of the volume control tank (VCT) during a LOOP? What volume of inventory is required in the VCT during a 4 hour LOOP? What inventory is normally stored in the VCT? Also, what are the setpoints for low VCT level alarm and for the closure of CHN-UV501, opening of CHN-UV514 and actuation of the BAM pumps, as shown on Figure 2 of your March 9, 1995, submittal? What is the probability that the VCT will drain down within 24 hours given a LOOP?

PVNGS Response 10:

There is no PRA success criteria for the VCT during a LOOP. The model assumes water is obtained from the refueling water tank (RWT), through the gravity feed path. Of the major chemical and volume control system (CVCS) design functions described in the UFSAR 9.3.4, only four functions apply during a LOOP. They are (1) maintain shutdown margin, (2) provide auxiliary spray, (3) provide RCS makeup during a cooldown, and (4) provide RCS makeup for small line breaks. All of these functions can be accomplished without the inventory contained in the VCT using equipment whose operability is controlled by technical specification requirements. Therefore, no inventory is required provided the tank can be isolated.

By procedure, the VCT level is normally controlled between 34% and 44% by an automatic makeup system. These levels correspond to an inventory of 1870 to 2280 gallons. However, since the tank automatically isolates at 5%, the "useful" volume of water in the tank is approximately 1180 to 1585 gallons.

A "VCT Level Hi-Lo" alarm is generated at a VCT level of 32%. This alarm gives the operator about 10 minutes to line up an alternate supply from the control room (i.e. gravity drain through CHN-HV536 with CHN-VU501 closed).

With no operator action a "VCT Level Lo-Lo" alarm will come in at a level of 5%. The associated switch will also generate signals to close CHN-UV501, open CHN-UV514, and start a BAM pump. The class powered valve CHN-HV536 is interlocked so that it will automatically open on a "VCT Level Lo-Lo" signal coincident with a "no power on CHN-UV514 valve" signal.

NRC Question 11:

The Engineering Evaluation of your March 9, 1995, submittal discussed the ability to recover from the charging pump gas binding event. If a charging pump gas binding event does occur, how long will it take operators to recover from this event? What is the probability that the operators will recover the charging pumps from a gas binding event as a function of time?

PVNGS Response 11:

If a charging pump becomes gas bound, the pump would be vented in accordance with applicable abnormal operating procedures. A qualified nuclear training instructor who has extensive experience in performing the evolution in administering the associated job performance measure for auxiliary operators indicates that the approximate time to perform the evolutions is between 45 and 60 minutes. The PRA does not credit recovery of charging (i.e., the probability of the operators not recovering charging is 1.0).