UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
611 RYAN PLAZA DRIVE, SUITE 400
ARLINGTON, TEXAS 76011-4005

April 8, 2005

EA-04-221

Gregg R. Overbeck, Senior Vice
President, Nuclear
Arizona Public Service Company
P.O. Box 52034
Phoenix, AZ 85072-2034

SUBJECT: FINAL SIGNIFICANCE DETERMINATION FOR A YELLOW FINDING AND NOTICE OF VIOLATION - NRC SPECIAL INSPECTION REPORT 2004-014 - PALO VERDE NUCLEAR GENERATING STATION

Dear Mr. Overbeck:

The NRC's January 5, 2005, inspection report described the results of a special inspection that followed up on your discovery in July 2004 that a significant section of containment sump safety injection piping at all three Palo Verde Nuclear Generating Station (PVNGS) units was void of water. The report discussed two findings that were being evaluated for further NRC action under the NRC's Significance Determination Process or NRC Enforcement Policy. This letter provides you the results of our evaluation of one of the findings, the preliminary "Greater than Green" finding involving a failure to maintain portions of the PVNGS emergency core cooling system (ECCS) filled with water in accordance with design control requirements. This finding was processed under the NRC's significance determination process. In separate correspondence, we are providing you the results of our enforcement deliberations on the second finding, an apparent violation of 10 CFR 50.59 that was processed under the NRC's Enforcement Policy.

NRC's evaluation of the design control finding considered the fact that Arizona Public Service Company (APS) discovered this condition at PVNGS in July 2004, following notification from another facility where a similar problem had been identified. On July 31, 2004, APS reported this condition to NRC under the provisions of 10 CFR 50.72(b)(3)(v), noting that the voided section of piping had the potential to prevent the fulfillment of the safety function to remove residual heat and mitigate the consequences of a loss-of-coolant accident. In early August, Palo Verde took corrective action to fill the voided piping in all three units, completing those actions by August 4, 2004.

At your request, a Regulatory and Predecisional Enforcement Conference was held on February 17, 2005, to discuss APS's perspectives on the risk significance of the design control issue. During the meeting the APS staff described their assessment of the significance of the finding, including the results of detailed pump testing APS sponsored to assess the performance of the high pressure safety injection (HPSI) and containment spray (CS) pumps with portions of the ECCS suction piping voided. The APS staff also described corrective actions, including the root cause evaluations for the failure to maintain the design of ECCS suction piping. The APS staff indicated that maintaining voided ECCS suction piping was contrary to the original design intent and was an unanalyzed condition. Your investigation identified possible causes as including: (1) the design requirement was specified, but the end user did not consider the design requirement and incorporate the requirement into procedures; (2) the design requirement was recognized, but there was a breakdown in communicating the design requirement to the end user; and (3) the design requirement was not recognized by the responsible design organization.

The APS staff indicated that the pump testing demonstrated that high pressure safety injection pumps would function for all loss of coolant accidents associated with a pipe break greater than 2.0 inches in diameter. Additionally, the APS staff indicated that, as a conservative measure during the significance determination, no change was made to your probabilistic safety assessment model to account for small-break loss of coolant accidents between 2.0 and 2.3 inches. The APS staff indicated that the significance of the finding should be characterized as having low to moderate safety significance (White) because the change in core damage frequency from the subject performance deficiency was 7.0×10^{-6}.

After considering the information developed during the inspection, the information APS provided at the conference, and the information APS provided in letters dated December 27, 2004, February 10, 2005, February 15, 2005, February 24, 2005, and February 28, 2005, the NRC has concluded that the inspection finding is most appropriately characterized as a Yellow finding, i.e., an issue with substantial importance to safety that will result in additional NRC inspection and potentially other NRC action. While we agreed with many of the assumptions that formed the basis for your risk determination, we identified many uncertainties associated with the conduct of the pump tests. A discussion of these uncertainties, their effect on our significance determination, and the primary basis for the NRC's conclusion, follows.

The NRC's review determined that the pump testing provided useful insights into post-accident high pressure safety injection pump performance. Nevertheless, there were several uncertainties associated with the testing and analysis methodologies that could have an impact on the overall conclusions regarding the availability of ECCS pumps following a loss-of-coolant accident. The significant test method uncertainties were in the areas of: (1) the use of the Froude Correlation and scaling, and (2) the impact of temperature on required net positive suction head. There were also several uncertainties associated with differences between the test configuration and the actual plant configuration. The significant configuration uncertainties were in the areas of: (1) the use of ambient temperature water during testing in lieu of post-accident temperature water, (2) the use of a method of air injection during the full scale testing that did not represent the actual void discovered in the plant, (3) the failure to model the transition between suction sources and the associated impact on check valve and system response, and (4) the failure during testing to account for post-accident conditions affecting the pump discharge.

We evaluated the above test method and test configuration concerns and concluded that they introduced large qualitative uncertainties associated with the selection of the loss-of-coolant accident break spectrum utilized by the APS staff in completing the safety analysis. After accounting for the uncertainties, we concluded that at least some portion of the medium loss-of-coolant accident break spectrum should be included in the significance determination of the failure to maintain the ECCS suction piping filled with water.

Taking into account these uncertainties, we determined that the most appropriate value for the change in core damage frequency lies between 5.7×10^{-6}, the result assuming that the performance deficiency only affects system response to small breaks, and 4.6×10^{-5}, the result assuming that high pressure safety injection pumps would fail on recirculation during a medium-break LOCA. Given that 89 percent of the range of core damage frequency lies in the Yellow region, as defined by the significance determination process, we have concluded that the most appropriate characterization of the significance of this finding is Yellow. Additional details of our evaluation and basis for arriving at a Yellow significance determination are contained in **Enclosure 2.**

We will use the NRC Action Matrix to determine the most appropriate NRC response for this issue. We will notify you by separate correspondence of that determination.

You have 30 calendar days from the date of this letter to appeal the staff's determination of significance for this Yellow finding. Such appeals will be considered to have merit only if they meet the criteria in NRC Inspection Manual Chapter 0609, Attachment 2.

The NRC also has determined that the failure to maintain portions of the Palo Verde ECCS in accordance with design specifications is a violation of 10 CFR Part 50, Appendix B, Criterion III, Design Control. This violation is cited in the enclosed Notice of Violation (Notice), **Enclosure 1**. The circumstances surrounding this violation were described in detail in the subject inspection report. In accordance with the NRC Enforcement Policy, the Notice of Violation is considered escalated enforcement action because it is associated with a Yellow finding. You are required to respond to the violation and should follow the instructions specified in the enclosed Notice in preparing your response.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. The NRC also includes significant enforcement actions on its Web site at www.nrc.gov; select **What We Do**, **Enforcement**, then **Significant Enforcement Actions**.

Sincerely,

/RA/

Bruce S. Mallett
Regional Administrator

Docket Nos. 50-528; 50-529; 50-530
License Nos. NPF-41; NPF-51; NPF-74

Enclosures: see next page

1. Notice of Violation
2. Final Significance Determination

cc w/Enclosures 1 & 2:

Steve Olea
Arizona Corporation Commission
1200 W. Washington Street
Phoenix, AZ 85007

Douglas K. Porter, Senior Counsel
Southern California Edison Company
Law Department, Generation Resources
P.O. Box 800
Rosemead, CA 91770

Chairman
Maricopa County Board of Supervisors
301 W. Jefferson, 10th Floor
Phoenix, AZ 85003

Aubrey V. Godwin, Director
Arizona Radiation Regulatory Agency
4814 South 40 Street
Phoenix, AZ 85040

M. Dwayne Carnes, Director
Regulatory Affairs/Nuclear Assurance
Palo Verde Nuclear Generating Station
Mail Station 7636
P.O. Box 52034
Phoenix, AZ 85072-2034

Hector R. Puente
Vice President, Power Generation
El Paso Electric Company
310 E. Palm Lane, Suite 310
Phoenix, AZ 85004

Jeffrey T. Weikert
Assistant General Counsel
El Paso Electric Company
Mail Location 167
123 W. Mills
El Paso, TX 79901

John W. Schumann
Los Angeles Department of Water & Power
Southern California Public Power Authority
P.O. Box 51111, Room 1255-C
Los Angeles, CA 90051-0100

John Taylor
Public Service Company of New Mexico
2401 Aztec NE, MS Z110
Albuquerque, NM 87107-4224

Cheryl Adams
Southern California Edison Company
5000 Pacific Coast Hwy. Bldg. DIN
San Clemente, CA 92672

Robert Henry
Salt River Project
6504 East Thomas Road
Scottsdale, AZ 85251

Brian Almon
Public Utility Commission
William B. Travis Building
P.O. Box 13326
1701 North Congress Avenue
Austin, TX 78701-3326

Enclosure 1

NOTICE OF VIOLATION

Arizona Public Service Company Palo Verde Nuclear Generating Station	Docket Nos. 50-528; 50-529; 50-530 License Nos. NPF-41; NPF-51; NPF-74 EA-04-221

During an NRC inspection completed December 8, 2004, a violation of NRC requirements was identified. In accordance with the "General Statement of Policy and Procedure for NRC Enforcement Actions," NUREG-1600, the violation is listed below:

> 10 CFR Part 50, Appendix B, Criterion III, Design Control states, in part, that measures shall be established to assure that the design basis is correctly translated into specifications, procedures, and instructions.
>
> The design basis for the Palo Verde Nuclear Generating Station (PVNGS) is specified, in part, in the plant Updated Final Safety Analysis Report (UFSAR). Section 6.3 of the UFSAR, "Emergency Core Cooling System," states, in part, that the safety injection piping will be maintained filled with water, and that during recirculation mode, the available net positive suction head for the containment spray and high pressure safety injection pumps is 25.8 feet and 28.8 feet, respectively (values that assume the pump suction piping is filled with water.)
>
> Contrary to the above, from initial plant licensing until July 2004, the design control measures established by the licensee were not adequate to assure that the design basis for the PVNGS emergency core cooling system (ECCS) was appropriately translated into specifications, procedures, and instructions. The licensee had no specifications, procedures or instructions in place to assure that the design basis for the ECCS system was maintained. Specifically, except for limited periods of time following ECCS leak testing prior to 1992, the licensee failed to maintain portions of the containment sump safety injection recirculation piping filled with water in accordance with the UFSAR, a nonconformance that affected the available net positive suction head for the containment spray and high pressure safety injection pumps as described in the UFSAR. This condition existed at Units 1, 2 and 3 of the PVNGS facility from initial plant operation (1985, 1986 and 1987, respectively) until August 2004, at which time corrective actions were taken to fill the affected piping.
>
> This violation is associated with a Yellow SDP finding.

Pursuant to the provisions of 10 CFR 2.201, Arizona Public Service Company is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555 with a copy to the Regional Administrator, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, TX, 76011, and a copy to the NRC Resident Inspector at the Palo Verde facility, within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation; EA-04-221" and should include for each violation: (1) the reason for the violation, or, if contested, the basis for disputing the violation or severity level, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken to avoid further violations, and (4) the date when full compliance will be achieved.

Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, One White Flint North, 11555 Rockville Pike, Rockville, MD 20852-2738. Because your response will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 8th day of April 2005

Enclosure 2

Final Significance Determination
Palo Verde Nuclear Generating Station
Voiding in the Recirculation Sump Suction Line

The NRC reviewed each of the differences between the licensee's evaluation documented in their analysis, Document Number 13-NS-C074, "Significance Determination of Containment Sump Air Entrainment," and presented during the Regulatory Conference held on February 17, 2005, and the NRC's preliminary significance determination. Using the additional data provided by the licensee, as well as evaluations and input from the NRC staff, a final significance determination was performed by modifying the preliminary evaluation as appropriate. The documentation that follows is not a stand-alone evaluation; the reader must also be familiar with the preliminary significance determination documented in NRC Special Inspection Report 05000528/2004014, 05000529/2004014, AND 05000530/2004014, Attachment D, "Phase 2 and Phase 3 Risk Assessments."

I. **Internal Events**:

Table 1 presents the differences between the licensee's evaluation and the NRC's preliminary evaluation for the internal initiators affected by the performance deficiency.

TABLE 1
Differences
Internal Events Assessment

Initiating Event	NRC Preliminary ΔCDF	PVNGS ΔCDF
Large LOCA	1.44×10^{-6}	0
Medium LOCA	1.06×10^{-5}	0
Small LOCA	9.15×10^{-7}	4.5×10^{-6}
Transients (PSV)	2.89×10^{-6}	2.7×10^{-7}
LOOP (RCP seal LOCA)	9.72×10^{-8}	0
Total	1.59×10^{-5}	4.8×10^{-6}

The NRC evaluated each of these differences, to determine the appropriate assumptions to use for the final determination. The following characterizes each of the changes made to the NRC's preliminary significance determination:

2. Recovery

Based on the testing performed, the licensee assumed that the containment spray pumps would not fail beyond nominal failure probabilities and that high pressure safety injection (HPSI) pumps would only fail during small-break loss-of-coolant accidents (LOCAs). In addition, because the failures were limited to only the most severe conditions on the pumps, the licensee assumed that the pumps would not be recoverable.

The NRC adjusted the preliminary analysis to indicate that only the HPSI pumps would fail and that this failure would damage the pumps such that they could not be recovered in the short term. Making these assumptions, the NRC used the Standardized Plant Analysis Risk Model for Palo Verde 1, 2, & 3 (SPAR), Revision 3.03 to quantify the change in core damage frequency (ΔCDF) related to small breaks. The change in assumptions resulted in an increase in ΔCDF to 3.41×10^{-6}.

3. Relief Valves

The licensee's probabilistic risk assessment model indicates that the pressurizer safety valves would not open if the auxiliary feedwater function is successful, and that they would always open if the auxiliary feedwater function is not successful.

The NRC agreed that the probability of a safety valve opening was dependent on the success or failure of auxiliary feedwater. Therefore, the NRC adjusted the SPAR output to indicate that the safety valves would always open during transients and special initiators if auxiliary feedwater was unavailable. However, the NRC did not agree that safety valves could never open given a success of the auxiliary feedwater function, but concurred that the value used in the SPAR was too high because the Palo Verde units do not have pilot-operated relief valves. Therefore, there are no actuation circuits that could inadvertently open the valves below setpoint and the safety valve setpoints at Palo Verde are significantly higher than the anticipatory pilot-operated relief valve setpoints at other plants modeled.

Therefore, the NRC used a screening value of 2.0×10^{-3} as the probability the safety valves open during a transient with successful auxiliary feedwater injection. This value was selected because it was an order of magnitude smaller than the nominal value used in the SPAR. Additionally, the NRC used the same assumptions used for the small-break LOCAs, namely, the containment spray pumps would not fail from the air in the suction lines and the HPSI pumps would fail upon recirculation and would not be recoverable.

The resulting ΔCDF related to transients and special initiators decreased to 2.33×10^{-8}.

4. <u>Reactor Coolant Pump Seals</u>

The licensee assumes that there is no possibility of a seal LOCA with their pumps, regardless of seal survival. The manufacturer of the seals has stated that the seal package clearances are small enough that a total failure of all seals would only result in a leak of 17 gallons per minute per pump. Therefore, there could never be a loss-of-coolant accident caused solely by the failure of the pump seals. The NRC determined that the total impact of this finding on loss of offsite power sequences were the result of seal LOCAs. As such, the finding had no impact on loss of offsite power initiators. Therefore, the NRC determined that the best estimate of ΔCDF was zero.

5. <u>Containment Spray Pumps</u>

The licensee assumed that the containment spray pumps would not fail at a greater rate as a result of the performance deficiency. They stated that their testing program proved that the containment spray pumps would continue to function throughout all accidents.

The staff determined that there were a significant number of concerns related to the applicability of the licensee's testing to the actual conditions that the pumps would be exposed to during an event. However, the NRC determined that most of these concerns would have a greater impact on the HPSI pumps. Therefore, the NRC assumed that the licensee's assumption was correct. This is a best estimate assumption based on a qualitative estimate of all inputs and data available. The resulting change was quantified with the changes discussed under Item 5.

6. <u>HPSI Pumps</u>

The licensee assumed that the HPSI pumps would function throughout all medium- and large-break scenarios. The licensee stated that their testing program proved that the finding would not prevent pump success for larger sized breaks because of the increased containment overpressure and the increase in flow rates through the pump. Therefore, the licensee concluded that there would be no increase in the core damage frequency for these accidents.

The staff concurred that the licensee's testing showed that as break size increased, the probability of failing the HPSI pumps decreased. Therefore, the NRC concluded that the licensee's assumption that the risk from large-break LOCAs was not affected by this finding was appropriate. The NRC determined that the best estimate of ΔCDF for large-break sequences was zero.

However, the staff had concerns with the scaling and applicability of the testing to indicate the ability of the HPSI pumps to perform their function in the plant. These concerns are documented in Section III of this document. Therefore, the NRC determined that the finding could result in a failure of the high pressure recirculation (HPR) function during a medium-break LOCA. As such, the NRC reviewed the range of potential changes in core damage frequency based on different assumptions. The value of ΔCDF, provided that the HPR function would never fail as a result of air in the sump lines during a medium-break LOCA, is zero. If air in the sump lines would have caused the pumps to fail throughout the entire range of medium-break sizes, the resulting ΔCDF would be 4.03×10^{-5} for medium-break sequences.

The concerns believed by the staff to have the most impact on how well the testing modeled actual plant conditions are documented in Section III, “Modeling and Scaling Concerns.”

II. External Events

Table 2 presents the differences between the licensee’s evaluation and the NRC’s preliminary evaluation for the external initiators affected by the performance deficiency.

TABLE 2
Differences
External Initiators Assessment

Initiating Event	NRC Preliminary ΔCDF	PVNGS ΔCDF
Seismic	7.90×10^{-6}	3.5×10^{-7}
Internal Floods	2.44×10^{-9}	1.0×10^{-8}
Internal Fire	9.26×10^{-7}	1.8×10^{-6}
Total External	8.83×10^{-6}	2.4×10^{-6}

The NRC evaluated each of these differences, to determine the appropriate assumptions to use for the final determination. The following characterizes each of the changes made to the NRC’s preliminary significance determination:

A. <u>Cooldown Following Seismic Event</u>

The major difference in the seismic analyses is that NRC did not give credit for cooldown following a seismic event as a success for a seismically induced small-break LOCA, as did the licensee. This was assumed because the atmospheric dump valves are not required safe shutdown equipment. Also, many additional activities would compete with the cooldown for operator attention given a catastrophic earthquake. The earthquake used by the NRC and the licensee was a seismic event three times larger than the design-basis earthquake. This event was given a likelihood, $F_{(3\text{-Seismic})}$, of 3.0 x 10^{-5}/yr in the Palo Verde Individual Plant Examination of External Events (IPEEE).

The NRC evaluated the licensee's assumptions. The IPEEE standard gave a deterministic requirement to review seismic events to ensure success paths following a seismically induced small-break LOCA. However, the licensee's seismic hazards evaluation determined that this event was unlikely. Therefore, the NRC used a screening value of 0.1 to model the likelihood that a seismic event would result in a small-break LOCA, $P_{(\text{SeismicLOCA})}$. Additionally, the NRC reviewed the sequences for a seismically-induced loss-of-coolant accident. Most of these sequences involved an extended period between event initiation and the receipt of a recirculation actuation signal. As such, the NRC determined that some credit should be given for operators cooling down the plant and placing it in shutdown cooling prior to the need for recirculation. Therefore, the NRC used a screening value of 0.1 for the basic event probability that operators failed to initiate a cooldown in time.

The conditional core damage probability, $CCDP_{(\text{SBLOOP})}$, for a seismically-induced small-break LOCA with consequential loss of offsite power while the performance deficiency existed was quantified by the SPAR as 1.454 x 10^{-1}. This value was dominated by the 10-percent probability that operators would not cool down the plant in a timely manner. The baseline conditional core damage probability, $CCDP_{(\text{Base})}$, for a seismically-induced small-break LOCA with consequential loss of offsite power was quantified by the SPAR as 2.764 x 10^{-2}. The resulting ΔCDF was calculated as follows:

$$\Delta CDF = F_{(3\text{-Seismic})} * P_{(\text{SeismicLOCA})} * CCDP_{(\text{SBLOOP})} - F_{(3\text{-Seismic})} * P_{(\text{SeismicLOCA})} * CCDP_{(\text{Base})}$$

$$= 3.0 \times 10^{-5}/\text{yr} * 0.1 * 1.454 \times 10^{-1} - 3.0 \times 10^{-5}/\text{yr} * 0.1 * 2.764 \times 10^{-2}$$

$$= 3.53 \times 10^{-7}/\text{yr} * 1 \text{ year} = 3.53 \times 10^{-7}$$

This represented a significant decrease in ΔCDF related to seismic initiators from the value presented in the preliminary significance determination.

B. Quantification of Internal Flooding

For internal floods, Palo Verde used their probabilistic risk assessment model loss of condenser vacuum and loss of nuclear cooling water event trees to quantify the change from internal risk. The NRC's preliminary estimate suggests that internal floods would pose no more risk than the internal events model.

The NRC adjusted the preliminary determination values to include the revised internal events assumptions and values from the SPAR. The resulting ΔCDF for internal flooding was increased to 7.90 x 10^{-9}. This value was very close to the licensee's quantification, and the exact value was not important to the final decision.

C. Quantification of Internal Fire

The NRC's preliminary and the licensee's fire analyses are essentially the same (less than a factor of two apart). Once the assumption changes in internal events were applied, the NRC's value was approximately 3 times higher than the licensee's determination. However, Palo Verde has a complete fire model in their probabilistic risk assessment. Therefore, the NRC determined that the best estimate would be the licensee's fully developed result of 1.80×10^{-6}.

III. Modeling and Scaling Concerns

The following issues were identified by the staff as the major concerns related to the modeling and scaling of the licensee's testing program. Each of these examples has the potential to affect the licensee's assumption that the finding does not impact the HPR function for reactor coolant system break sizes in the medium-break range.

4. Froude Correlation:

The use of Froude numbers as the dimensionless parameter for scaling the test model was questioned. Actual test results indicated that the Froude number was not the best scaling factor for the vertical downcomer. The licensee changed the scaling of this pipe as a result. The staff questioned the use of the Froude number to model the section of piping from the downcomer to the bifurcation heading to the HPSI pump suction. If the ratio of the bubble rise rate to actual pipe velocity was not properly characterized, the result could be a significantly higher air ingestion rate for the HPSI pumps.

The staff reviewed additional information provided by the licensee on February 15, 2005, in response to questions raised at the regulatory conference. One example of issues identified during this review was that the licensee provided a document used to justify their use of Froude number scaling in modeling large pipes (United States Department of the Interior - Engineering Monograph No. 41, "Air-Water Flow in Hydraulic Structures"). This document raised the following additional problems or concerns:

- For closed pipe, especially with partial fill and elevation changes, the report used a different Froude number equation than in the Palo Verde model.

- In discussing partially filled closed systems it stated, "There are many literature references that indicate model predictions often underestimate in the quantity of air which actually flows in prototype structures."

- The document stated, "studies clearly indicate that for estimating airflow rates using models, it is necessary to accurately reproduce the entire airflow passage above the water" (meaning not averaging values). The calculations for this were extremely complex "summation" equations with the Froude number being just one of many variables.

- The document stated, ".... conditions can exist whereby bubbles will move downstream and form into pockets that move against the flow in an upstream direction. Studies investigated prototype cases in which large air pockets moved against the flow with sufficient velocity to completely destroy reinforced concrete." This indicated to the NRC staff that a water hammer analysis was necessary.

- Finally, the document described complex variations in the air-water flow regimes based on the flow devices (valves, gates, etc.) and the timing of these devices. The staff was concerned that the valve types, initial conditions, and timing of the model could have a major impact on the results.

2. Modeling of Pump Discharge

The licensee's full-scale model did not have a check valve nor did it model the continuous back pressure that would be provided by the reactor coolant system. As a result, the staff assumed that the test was much more applicable to steady state conditions and did not model the likelihood that air ingestion would cause intermittent closure of the discharge check valve and increase the likelihood that the pump would air bind.

3. Transition Between Suction Sources

The licensee did not model the switch over from the refueling water tank to the containment sump during the scale model testing. Additionally, the licensee did not provide a complete assessment of the differences that may ensue between the test configuration and that of the plant from the switch over dynamics. The staff is concerned that the timing of various valve actions would have perturbed the behavior of gas and water movement and may have introduced an unrecognized error in the assessment.

4. Air Injection Near Pump Suction

The licensee's full-scale test involved injecting air into an established steady flow close to the pump inlet. The staff believes that this model would not have the same effect on the net positive suction head available as it would if the air were distributed in the entire flow. Also the established steady flow and flow momentum would help to sweep air through the pump.

The presence of a high void fraction in the vertical section of the suction pipe would decrease the net positive suction head available while significantly increasing net positive suction head required. Inadequate net positive suction head can cause pump cavitation, pump vibration, and even pump failure. In the mockup test, big fluctuations in flow and pressure were observed. The fluctuations could have been magnified by the presence of a large amount of air in the entire suction flow region.

5. Containment Environment/Temperature Concerns

The simulated containment pressure used by the licensee in the full-scale mockup was not the same as the full range of containment pressures under LOCA conditions. Also, the containment pressure for the mockup was not consistent with the sump water temperature. The use of 4.2 psig containment pressure was not consistent with the 70-degree temperature fluid that was used in the test. Based on accident analysis, a 4.2-psig containment pressure would correlate to a much higher sump water temperature. Use of 70 degrees F water while maintaining containment pressure at 4.2 psig is nonconservative with respect to pump performance.

IV. Sensitivities

The NRC evaluated the sensitivity of the preliminary significance determination results to each of the assumptions that differed from the licensee's evaluation assumptions. For each of the assumptions documented in Sections I and II above, the NRC modified the preliminary results to reflect the licensee's assumption. The result, using the NRC's original quantification techniques and tools, was sufficiently close to the licensee's evaluation result that it indicated that all important assumption differences were represented. The adjusted preliminary significance determination result, provided in Table 3, was derived using the original calculational techniques while making changes that resulted in the assumptions listed below. Those assumptions that were accepted as the NRC's best estimate assumptions are indicated as "best estimate." Those assumptions where significant uncertainties existed related to the licensee's evaluation are marked as "uncertain." The "uncertain" assumptions were used in the adjustment solely to provide a baseline for evaluation of the sensitivity of each assumption.

< The containment spray and low-pressure safety injection pump performance was not affected by the subject finding (uncertain).

< The HPSI pumps were only affected during small-break LOCA sequences (uncertain).

< Upon failure, the HPSI pumps will not be recoverable because only the worst impact sequences are being evaluated (best estimate).

< Primary safeties will always lift upon loss of the auxiliary feedwater function, but will never lift if auxiliary feedwater is successful (best estimate).

< A loss of cooling to the reactor coolant pump seals can never cause a loss-of-coolant accident (best estimate).

< Ten percent of seismic events of a magnitude larger than 3 times the design basis will result in a small-break LOCA (best estimate, bounding).

< Operators will fail to cooldown the reactor prior to recirculation 10 percent of the time following a seismically-induced LOCA (best estimate, bounding).

< The result from the licensee's fire PRA is the best available estimate of change in core damage frequency related to the finding related to internal fires (best estimate).

The NRC challenged each of these assumptions by restoring the value associated with the assumption to that used in the preliminary significance determination. The resulting ΔCDF for all initiators following the change in each specific assumption is presented in Table 3.

TABLE 3
Sensitivities
Critical Assumptions Evaluated

Assumption	Change Evaluated	Result (ΔCDF)
Adjusted Preliminary Significance	N/A*	6.7×10^{-6}
HPR Always Affected	CCF set to 1.0 for MBLOCA	4.7×10^{-5}
CSR Fails with HPR	CSR Set to 1.0	1.7×10^{-5}
Primary Safeties Fail at Nominal	Open in 2% of all Transients	1.6×10^{-5}
Primary Safeties Fail with AFW	Always Open on Loss of AFW	5.7×10^{-6}
RCP Seal LOCA	LOCA occurs on SBO	1.9×10^{-5}
HPR Affected During LBLOCA	CCF set to 1.0 for all LOCAs	4.7×10^{-5}
Recovery	HPR Recovered 76% of time	1.6×10^{-6}
Cooldown following Seismic	No Cooldown	9.3×10^{-6}

* NOTE: The NRC's adjustment included the differences stated above and assumptions considered to have a high degree of uncertainty, and should not be misinterpreted as a best estimate evaluation.

Abbreviations Used in Table 3:

- HPR: High Pressure Recirculation
- N/A: Not Applicable
- CCF: Common Cause Failure
- CSR: Containment Spray Recirculation
- AFW: Auxiliary Feedwater
- RCP: Reactor Coolant Pump
- LOCA: Loss-of-Coolant Accident
- SBO: Station Blackout
- LBLOCA: Large-Break LOCA

The NRC reviewed the results from the sensitivity study. None of the results indicated that the significance of the subject finding should be classified below WHITE. While several of the assumption changes indicated that the significance of the finding would best be characterized as YELLOW, all but one assumption resulted in a significance result that was at or near the WHITE/YELLOW boundary.

The only assumption that had an appreciable change (greater than a factor of 2 - 3) in the significance was whether the performance of the HPSI pumps would be impacted during a medium-break LOCA.

V. Final Significance Determination

The NRC made changes to the preliminary analysis for all assumption differences documented in Sections I and II, that the staff agreed were appropriate to use in a best estimate evaluation. While several assumptions were questioned by the staff, the NRC determined that most assumptions did not affect the final significance regardless of the assumption used. Therefore, the NRC reevaluated the significance of the subject finding by revising the preliminary significance determination using the licensee's assumptions. The result of these adjustments is reflected in Table 4, Column 3.

The NRC then determined by reviewing the sensitivities documented in Table 3, that the only assumption that could impact the final significance determination was the licensee's assumption that the finding would not affect medium-break LOCA sequences. The NRC revised the evaluation documented in Table 4, Column 3 by assuming that the entire spectrum of medium-break LOCAs would be affected by the performance deficiency. Specifically, the NRC assumed that the HPR function would fail during LOCAs falling anywhere within the entire spectrum of small and medium breaks. The result of this adjustment is reflected in Table 4, Column 4.

TABLE 4 Range of Final Significant Determination Internal Events Assessment			
Column 1 Initiating Event	**Column 2 Preliminary ΔCDF**	**Column 3 Adjusted No MBLOCA Failures ΔCDF**	**Column 4 Adjusted w/ MBLOCA Failures ΔCDF**
Large LOCA	1.44×10^{-6}	0	0
Medium LOCA	1.06×10^{-5}	0	4.03×10^{-5}
Small LOCA	9.15×10^{-7}	3.41×10^{-6}	3.41×10^{-6}
Transients (PSV)	2.89×10^{-6}	2.33×10^{-8}	2.33×10^{-8}
LOOP (RCP seal LOCA)	9.72×10^{-8}	0	0
Seismic	7.90×10^{-6}	4.36×10^{-7}	4.36×10^{-7}
Internal Floods	2.44×10^{-9}	7.90×10^{-9}	7.90×10^{-9}
Internal Fire	9.26×10^{-7}	1.80×10^{-6}	1.80×10^{-6}
Total	**2.47×10^{-5}**	**5.68×10^{-6}**	**4.60×10^{-5}**

Given these evaluations, the NRC determined that the best estimate value for the change in core damage frequency lies somewhere between the 5.7×10^{-6}, the result using the licensee's assumptions, and 4.6×10^{-5}, the result using those assumptions determined to be the best available information by the NRC and the preliminary significance determination assumption that HPSI pumps would fail on recirculation during a medium-break LOCA. The NRC noted that, given this uncertainty range, approximately 89 percent of the total range lies in the Yellow region defined by the significance determination process.

Based on the uncertainties associated with the phenomena involved, and the non-conservatisms and other issues associated with the testing documented in Section III, the staff could not conclude, with reasonable assurance, that there was a high likelihood that one or both HPSI pumps would have performed its risk-significant function during the postulated scenarios. The NRC noted that the probability of having a medium-break LOCA within the spectrum of breaks decreases greatly with the size of the break. This indicates that the majority of the initiating event frequency is the probability of having a medium-break LOCA that is on the low end of the break-size spectrum. Therefore, relatively small errors in the licensee's method of determining the maximum break size that would fail the HPR function could have a large effect on the final result of the risk determination.

The NRC calculated that an increase in the common cause failure rate of the HPR function, caused by the subject performance deficiency, of between 8 and 14% (depending on the model used) throughout the spectrum of medium breaks, would represent the White/Yellow threshold. Based on the NRC's determination that the pumps would be impacted by air intrusion and would not survive during these scenarios, the NRC determined that the subject performance deficiency is more appropriately characterized as Yellow, a finding of substantial safety significance.

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