Palo Verde Nuclear Generating Station

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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
Redacted Version of Proprietary Submittal Dated February 15, 2005 Regarding Safety Significance Evaluation of ECCS Containment Sump Voided Piping

In letter no. 102-05213, dated February 15, 2005, Arizona Public Service Company (APS) submitted to the NRC the safety significance evaluation of emergency core cooling system (ECCS) containment sump voided piping. APS requested that Enclosure 2 and Attachments 2-A, 2-B, 2-C, 2-D, and 2-E of that submittal be withheld from public disclosure under 10 CFR 2.390(a)(4) because they contained information considered to be proprietary to APS. Since that time, NRC Region IV personnel have requested that APS submit redacted versions of Enclosure 2 and Attachments 2-A, 2-B, 2-C, 2-D, and 2-E of the February 15, 2005 submittal. The requested redacted versions of the enclosure and attachments are enclosed.

There are no commitments in this letter. Should you have any questions, please contact Mr. Thomas N. Weber at (623) 393-5764.

Sincerely,

Gregg R. Overbeck

GRO/TNW/GAM/ca

Enclosure: Redacted Versions of Proprietary Enclosure 2 and Attachments 2-A, 2-B, 2-C, 2-D, and 2-E of APS Letter No. 102-05213, dated February 15, 2005, Regarding Safety Significance Evaluation of ECCS Containment Sump Voided Piping

cc:			
	T. W. Pruett	NRC Region IV	(w/ Enclosure)
	B. S. Mallett	NRC Region IV Regional Administrator	(w/o Enclosure)
	M. B. Fields	NRC NRR Project Manager	"
	G. G. Warnick	NRC Senior Resident Inspector for PVNGS	"

Redacted Versions of Proprietary Enclosure 2 and Attachments 2-A, 2-B, 2-C, 2-D, and 2-E of APS Letter No. 102-05213, dated February 15, 2005, Regarding Safety Significance Evaluation of ECCS Containment Sump Voided Piping

ENCLOSURE 2

SAFETY SIGNIFICANCE EVALUATION OF ECCS CONTAINMENT SUMP VOIDED PIPING

(Proprietary)

REDACTED VERSION

SIGNIFICANT CRDR 2726509
SAFETY SIGNIFICANCE EVALUATION OF ECCS CONTAINMENT SUMP VOIDED PIPING

Executive Summary

In July, 2004, Engineering personnel determined that a section of Emergency Core Cooling System (ECCS) piping leading from the containment recirculation sump, in both ECCS trains in each of the three Palo Verde Units, was left in an unfilled condition during normal plant operation. The resultant volume of air could potentially be ingested into the ECCS pumps suction following a Recirculation Actuation Signal (RAS). A review of design basis information determined that this condition was not consistent with the design intent of the ECCS and not consistent with the analyses that demonstrate the ability of the ECCS to perform its design basis safety functions. Condition Report/Disposition Request (CRDR) 2726509 was initiated to document and evaluate the condition.

The purpose of this report is to describe and provide the results of a comprehensive testing and analysis program performed to evaluate the ECCS system response to the voided piping condition. The results of the evaluation are then used in a risk assessment to determine the safety significance of the discovered condition.

Scale model tests were performed at Fauske and Associates which simulated the system response during and following a RAS with the affected section of piping initially unfilled. The scale tests were conducted in phases. The purpose of the first phase (typically referred to as Phase 1) was to demonstrate the ability to simulate the transient and measure the important parameters such as void fraction, pressure, and flow rate. [

]

[ ]

Full-scale pump tests were performed [ ] to determine the impact on pump performance under the projected air ingestion conditions. [

]

[ ]

[

]

In addition to the testing program, a computer hydraulic transient analysis of the ECCS voided pipe condition was performed. [

Ultimately, the analysis results are compared to the testing program and shown to be complimentary.]

Given the results of the tests and analyses, the risk significance was determined by making appropriate adjustments to the Palo Verde Probabilistic Risk Assessment (PRA) model. [

]

1 Introduction

1.1 Background/Purpose of Report

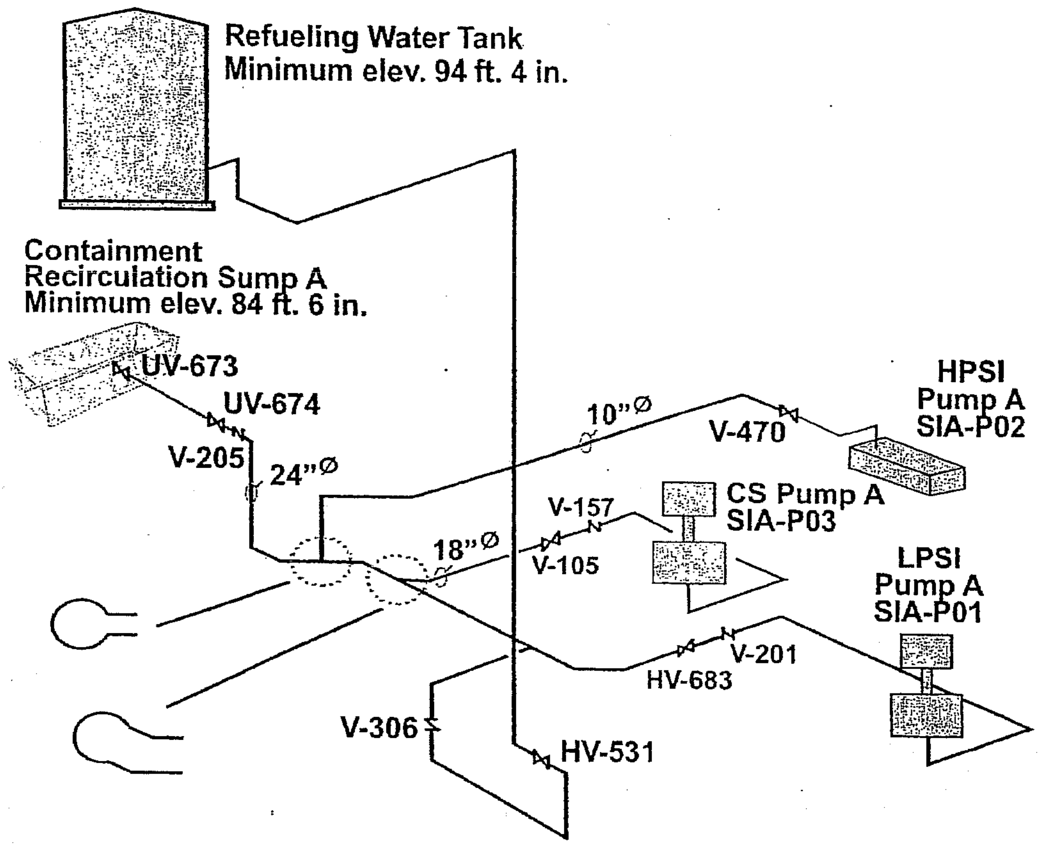
In July, 2004, Engineering personnel determined that a section of Emergency Core Cooling System (ECCS) piping leading from the containment recirculation sump, in both ECCS trains in each of the three Palo Verde Units, was left in an unfilled condition during normal plant operation. The resultant volume of air could potentially be ingested into the ECCS pumps suction following a Recirculation Actuation Signal (RAS). A review of design basis information determined that this condition was not consistent with the design intent of the ECCS and not consistent with the analyses that demonstrate the ability of the ECCS to perform its design basis safety functions. Condition Report/Disposition Request (CRDR) 2726509 was initiated to document and evaluate the condition.

The purpose of this report is to describe and provide the results of a comprehensive testing and analysis program performed to evaluate the ECCS response to the voided piping condition. The results of the evaluation are then used in a risk assessment to determine the safety significance of the discovered condition.

1.2 Description of Condition

The Palo Verde ECCS design employs recirculation from the containment sump after the contents of the Refueling Water Tank (RWT) have been injected into the reactor vessel and containment building. Upon receipt of a RAS, automatic valve actuations result in suction of the ECCS pumps being transferred from the RWT to the containment sumps. Two completely redundant and separated ECCS trains are utilized. Figure 1-1 illustrates a typical ECCS suction piping and component layout.

Emergency Core Cooling and Containment Spray System Suction Piping - Train A



-Not to scale-

Figure 1-1 Typical Palo Verde ECCS Suction Layout

As illustrated in Figure 1-1, the containment sump outlet pipe contains an in-board and an out-board containment isolation valve, and a downstream check valve. Engineering personnel determined that this section of the ECCS suction piping, between the two containment isolation valves and between the out-board valve and the downstream check valve, had been routinely left in an unfilled condition during plant operation.

In the unlikely event of a Loss-of-Coolant Accident (LOCA), the contents of the Reactor Coolant System (RCS) will leak into containment and flow into the containment sumps. Automatic ECCS actuation would occur causing the contents of the RWT to be injected into the RCS and the containment building to maintain core cooling and containment pressure and temperature control. Ultimately the basement of the containment building, including the containment sumps, would become flooded. Once the contents of the RWT are depleted, a RAS would be automatically generated causing both containment sump isolation valves in each train to open, resulting in closure of the RWT isolation check valves. The RAS would also cause, by design, the Low Pressure Safety Injection (LPSI) pumps to be turned off. ECCS suction, consisting of a HPSI pump and a CS pump in each train, would thus be transferred to the containment sump.

With the containment sumps flooded and the section of containment sump piping not filled with water, air would be trapped in the piping. As flow is initiated from the sump, this air could be entrained and/or transported into the ECCS suction piping and potentially into the ECCS pump inlets. Industry literature and operating experience indicates that pump performance could be severely degraded, or even result in air binding or pump failure, if the resultant air volume fraction ingested by the pump exceeds the pump's tolerance for air ingestion. Industry literature (Ref. 1 NUREG/CR 2792) indicates that a pump's tolerance for air ingestion varies by design and fluid conditions, but at air volume fractions above approximately 3%, pump degradation can be experienced.

Therefore, in order to determine the safety significance of this condition, the air volume fraction that could be ingested by the HPSI and CS pumps would need to be determined. Once the air volume fraction is determined, each pump's tolerance for the projected air ingestion can be assessed, and ultimately the impact on the ECCS safety functions.

1.3 Significance Determination Approach

The assessment of voided and two-phase fluid behavior is complex. A comprehensive scale model testing program was employed to develop a full understanding of the system response to the void and the resulting air/fluid conditions that would be delivered to the pumps' suction inlet. The impact to pump performance was then assessed via full-scale testing, given the projected air/fluid inlet conditions.

The scale model tests were performed at Fauske and Associates, and simulated the system response during and following a RAS with the affected section of piping initially voided. The scaled tests were conducted in phases. The first phase modeled the RWT and associated piping, and the sump and associated piping down through and including the long vertical run of pipe. The purpose of the first phase (typically referred to as Phase 1) was to demonstrate the ability to simulate the transient and measure the important parameters such as void fraction, pressure, and flow rate. A series of tests were performed to test important scaling parameters to ensure the results of the test could be confidently applied to the full scale Palo Verde units. A series of phenomenological tests using a larger scale model was incorporated into the test plan to verify that the flow regime in the vertical section of the scaled piping configuration was representative of large pipe behavior.

The second phase extended the scale model to include the individual pump suction piping up to each pump inlet. An extensive series of tests under varying flow and pressure conditions were performed. [

] These results established the inlet conditions for the subsequent full-scale pump performance tests.

Full-scale pump performance tests were performed at Wyle Labs utilizing a spare Palo Verde High Pressure Safety Injection (HPSI) pump and a representative Containment Spray (CS) pump to determine the impact on pump performance under the projected air ingestion conditions. The HPSI pump was of the same make and model as those installed at Palo Verde. A spare CS pump of the same make and model as the Palo Verde CS pumps was not readily available; therefore a spare CS pump from a cancelled WPSS plant was utilized for the test. This pump is the same make and model as the Palo Verde LPSI pumps and is very similar in design and size to the Palo Verde CS pumps. The impact on performance for equivalent fluid conditions is expected to be representative. Tests were performed for a spectrum of flow rates and air ingestion rates based on the results of the scale model test program. Pump performance was measured as a function of air volume fraction. A maximum degraded pump performance curve was then constructed using the test results for the tests performed at maximum air volume fractions.

[ ]

[ ] For those system conditions in which the required head do not exceed the degraded pump performance capability, continued degraded ECCS delivery (i.e. continued pump flow) is assumed until the air inventory available for ingestion into the pump is consumed, at which time restoration of full pump performance is assumed.

[ ]

[ ]

2 Scale Model Testing

2.1 Phase 1 Test Program and Results

2.1.1 Experimental Objectives and Physical Arrangement

The objective of the Phase 1 testing was to investigate the potential for the air initially resident in the horizontal piping section from the containment sump to be forced into the vertical downward piping section. Phase 1 tests included the transient effects of switching the supply from the simulated RWT to the simulated containment sump by simultaneously opening the sump suction isolation valves. Clear piping was used for the horizontal and vertical segments of the simulated suction line to observe and record the flow pattern and the behavior of the initial air filled void. A complete report on the conduct and results of the Phase 1 test program is attached as Attachment 2-A to this report.

The test facility that was used was comprised of two tanks with water inventories, a centrifugal pump, piping, valves, and associated instrumentation. The piping and valves used to establish and visualize the flow pattern development from the initial location between the valves and into the downcomer piping were all 4 inch in diameter. Clear plastic piping facilitated observation of the initial air inventory behavior during the opening of the motor operated valves. The vertical segment was also clear plastic piping that allowed for the observation [ ] in the downward vertical flow. [ ]

2.1.2 Scaling Considerations

As indicated, 4 inch diameter piping was used to simulate the sump horizontal and vertical downward sections of piping. Since actual Palo Verde piping is 24 inch in diameter, this results in a $1/6^{th}$ geometric scaling factor. This geometric (lengths and diameters) scaling factor was maintained through out the Phase 1 tests to the extent possible.

Previous tests and experiments described in the literature have demonstrated that maintenance of the Froude number, particularly for horizontal flow regimes, will result in prototypical behavior in scaled experiments. As such, flow rates were scaled in the Phase 1 tests so as to maintain the same dimensionless Froude Number parameter as would exist in the Palo Verde units.

2.1.3 Phase 1 Results and Observations

A series of twelve tests were performed with varied [ ]

[ ]

2.2 Phenomenological Testing Program

2.2.1 Experimental Objective and Physical Arrangement

Design reviews conducted before and after the Phase 1 tests and an independent review [ ] resulted in the identification of several phenomenological investigations that could be performed to provide

[ ]

The test arrangement also provided the opportunity to observe the flow patterns and influence of the HPSI and CS branch connections off the lower header piping.

2.2.2 Phenomenological Testing Results and Observations

An extensive series of tests using the [ ] scale test apparatus were performed. Key observations from these tests were

[ ]

2.3 Phase 2 Test Program and Results

2.3.1 Experimental Objectives and Physical Arrangement

The test facility for Phase 2 was similar to that of Phase 1[

]

[

]

Figure 2-1 Phase 2 Test Arrangement.

[

] In the plant system under accident conditions, air transported through the HPSI line would influence the pump performance and cause a decrease in the flow rate being pumped. Reduced flow rate would cause a corresponding reduction in the rate of air ingestion. Thus, the air intrusion rate deduced from these scaled experiments provides a conservative representation of the plant response.

The test instrumentation is also illustrated in Figure 2-1. A computer with a CIO-DAS008 data acquisition card was used to collect the data. Key pieces of instrumentation included

[ ]

- Various pressure , level, and flow meters

[ ]

During the Phase 2 tests, the flow rate through the CS pump was again held constant at the maximum predicted flow rate equivalent to 4885 gpm, except for several tests in which CS flow was set to zero to simulate a HPSI flow only scenario. HPSI flow rate was varied ranging from the equivalent to 200 gpm to an equivalent maximum run-out flow of 1310 gpm. [
]

[ ]

2.3.2 Scaling Considerations

The same 1/6th geometric scaling used in Phase 1 was used for the Phase 2 experiments. Flow rates were scaled to maintain the same Froude number that would exist at Palo Verde. The Froude number relationship was maintained for both the total flow and the individual flow rates to the simulated HPSI, CS, and LPSI pumps.

[ ]

[ ]

In this horizontal orientation, the principal scaling parameter has been well established previously (References 3 and 4) to be the Froude number which is a ratio of the inertial and buoyancy forces, i.e.

$$N_{Fr}^2 = \frac{\rho_w U^2}{gD\left(\rho_w - \rho_g\right)} \qquad \text{Eq. (1)}$$

where:

- D is the diameter of the horizontal piping,
- g is the acceleration of gravity,
- U is the one-dimensional velocity of the flow in this line,
- ρ_g is the air density, and
- ρ_w is the water density.

Since $\rho_w >> \rho_g$, this reduces to the familiar form

$$N_{Fr} = \frac{U}{\sqrt{gD}} \qquad \text{Eq. (2)}$$

[ ]

2.3.3 Phase 2 Results and Observations

A series of twenty-eight tests were initially performed with varied flow rates, containment level, and containment pressure conditions. Additional tests were later performed to investigate the air transport process during potential LPSI pump start scenarios. Key observations from the tests were:

Flow Patterns

Digital movie cameras were used to record the flow patterns in all the Phase 2 tests. Each test was initiated by simultaneously opening the sump containment isolation valves. As the valves open, water is seen to enter the initially voided horizontal piping segment and induce mixing of the water and air. The air is swept out of the horizontal segment and into the vertical piping segment. [

]

HPSI Air Ingestion Rates

[

]

[ ]

[ ]

These results show that the air flow ingestion rates increase to their maximum value within approximately [ ] seconds for the scaled experiments and then subsequently decay towards zero as the air inventory in the horizontal suction header becomes insufficient to enter the HPSI line. Similar evaluations for scaled HPSI flow rates [ ]

[ ]

[ ]

With a 1/6th linear scale, the respective volumes are determined by the cube of this linear scale, i.e. the scaled up quantities are defined by the volume multiplied by 216. More simply put, the area is scaled by the square of the diameter times the length. Thus six cubed equals 216. Since mass is directly proportional to volume at a given pressure and temperature, mass quantities are also scaled by a factor of 216.

[ ]

[ ]

Using the results from the Phase 2 tests, these scale factors are applied and the results illustrated in Figure 2-4 for the case of a HPSI flow rate of 1310 gpm. As shown, the meaningful delivery period for the air flow is approximately [ ]

[

]

Since Reference 1, and other pump performance tests described in the literature, indicates that pump performance is typically assessed as a function of air volume fraction, the peak mass flow rate data obtained during the Phase 2 tests was converted to air volume fractions for use in the full-scale pump tests. [

]

[ ]

3 Hydraulic Transient Analysis

3.1 Description of Analysis and Computer Model

A hydraulic computer model of a typical Palo Verde ECCS system was developed [

]

3.2 Analysis Results

[ ]

[ ]

[ ]

3.3 Hydraulic Transient Analysis Conclusions

[ ]

[ ]

4 Pump Performance Testing With Air Ingestion

4.1 Description of Test Facility

The pump performance tests were conducted at Wyle Labs in Huntsville, AL. The test facility consisted of two closed pump loops each drawing suction from, and discharging to, a common 30,000 gallon pressure vessel. One loop was constructed to provide for testing of the spare HPSI pump. Suction and discharge pipe sizes were selected to correspond to the actual pipe sizes at Palo Verde. The specific suction piping configuration leading into the HPSI suction nozzle was explicitly reproduced. The second loop was provided for testing of the representative CS pump.

[ ]

4.2 Test Conduct

[ ]

For each base case, tests were performed at incrementally increasing air injection mass flow rates. The resulting air volume fraction, defined as the ratio of volumetric air flow rate to total volumetric air flow rate, was then determined. [ ] Figure 4-1 illustrates the final test for the [ ] base case. [ ]

[ ]

Figure 4-1 Air Injection and Air Volume Fraction for Final 900 gpm Series Test

During every test, the duration of air injection was specified to assure that the total volume of air [ ] exceeded the total volume of air predicted by the scale model tests. Pump performance data was taken during each test for subsequent assessment of the air ingestion on pump performance. Visual observations, and digital camera recordings, were made for all HPSI test cases.

4.3 Test Results

Visual observations through the clear spool piece on the HPSI suction line confirmed [ ] similar in nature to that observed during the scale model Phase 2 tests. The visual observations confirmed the proper scaling of the Phase 2 tests and gives reasonable confidence that the Phase 2 and Phase 3 tests closely approximate the full-scale plant conditions. Pump performance data was taken using a data acquisition system that recorded each data point 10 times per second. The recorded data was then inserted into Excel spreadsheets to facilitate calculation of pump developed [ ] The data represents the calculated developed head (TDH) from the recorded pump inlet and outlet pressure data taken every 0.1 seconds, and the corresponding flow rates as measured on the pump discharge line. The data represents that obtained over a specific time period during which the air injection rate was at its maximum steady state value and the corresponding peak air volume fractions were obtained. The data points, as expected, fall along the test loop system curve.

As illustrated in the preceding three figures, and as would be expected, pump performance progressively degrades as inlet air volume fraction increases. This progressive degradation is consistent with data reported in NUREG/CR 2792 (Reference 1). The following figure 4-5 is taken from Reference 32 as cited in the NUREG.

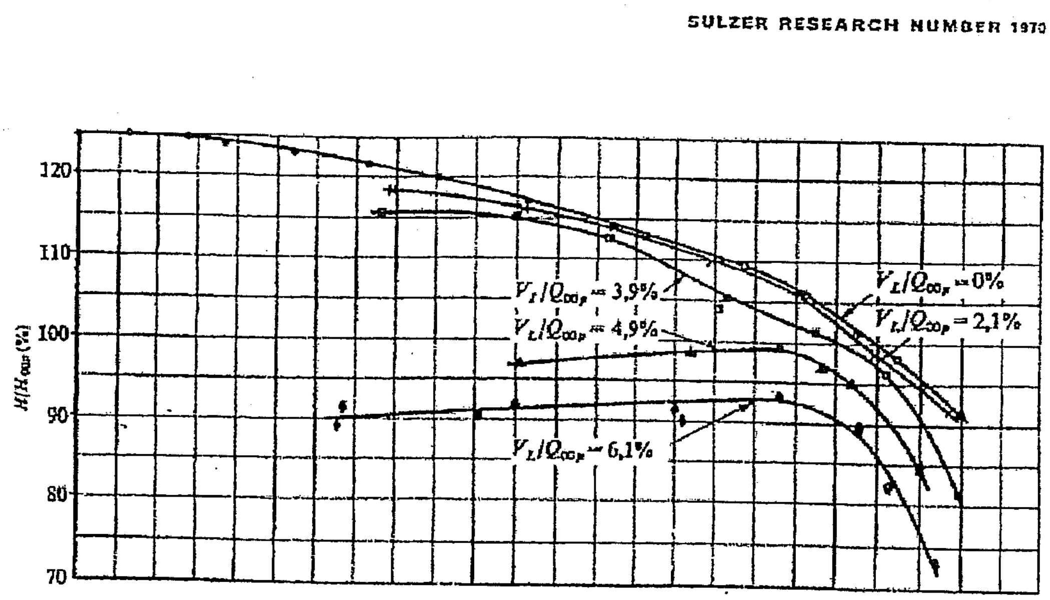


Figure 4-5 Degrading Pump Performance as a Function of Air Volume Fraction

A maximum bounding degraded pump curve is then constructed as shown in Figure 4-6. As illustrated, the maximum degraded pump curve conservatively bounds all recorded data for the peak air volume fraction cases tested. The use of this maximum degraded pump curve results in additional conservatism since the Phase 3 tests conditions in some cases exceeded the specified air volume fraction from the Phase 2 scale model tests.

[redacted]

5 Safety Function Impact

5.1 Thermal Hydraulic Analysis of Spectrum of LOCA Break sizes

A series of thermal hydraulic analyses of the Palo Verde ECCS system were performed using the [ ] These analyses established the expected [ ] conditions that would exist at the time of RAS for a spectrum of LOCA break sizes. Operator actions as prescribed in the Palo Verde Emergency Operating Procedures (EOPs) to initiate a cool down and depressurization of the RCS upon diagnosis of a LOCA were explicitly considered in the analyses. In this way, best-estimate parameters [ ] at time of RAS were established. [ ] descriptions of the [ ] codes are presented, followed by [ ] Detailed descriptions of the codes and their applications and limitations are within References [ ] These references also provide detailed descriptions of the individual transient results.

5.1.1 MAAP4 Analysis Code Description

MAAP is a computer code that simulates light water reactor system response to accident initiation events. The Modular Accident Analysis Program (MAAP), an integral systems analysis computer code for assessing severe accidents, was initially developed during the industry-sponsored IDCOR Program. At the completion of IDCOR, ownership of MAAP was transferred to Electric Power Research Institute (EPRI). Subsequently, the code evolved into a major analytical tool (MAAP 3B) for supporting the plant-specific Individual Plant Examinations (IPEs) requested by NRC Generic Letter 88-20. Furthermore, MAAP 3B was used as the basis to model the Ontario Hydro CANDU designs. As the attention of plant-specific analyses was expanded to include accident management evaluations, the scope of MAAP (its design basis) was expanded to include the necessary models for accident management assessments. MAAP4 is the first archived code that contains a graphical representation of the reactor and containment response. MAAP4, like MAAP 3B, is currently being maintained by Fauske & Associates, LLC (FAI) for EPRI and the MAAP User's Group (MUG).

MAAP4 is an accident analysis code that provides results with confidence in all phases of severe accident studies, including accident management, for current PWR reactor/containment designs and for ALWRs. MAAP4 includes models for the important accident phenomena that might occur within the primary system, in the containment, and/or in the auxiliary/reactor building. For a specified reactor and containment system, MAAP4 calculates the progression of the postulated accident sequence, including the disposition of the fission products, from a set of initiating events to either a safe, stable state or to an impaired containment condition (by overpressure or over-temperature) and the possible release of fission products to the environment.

Since the beginning of the MAAP code development, the codes have represented all of the important safety systems such as emergency core cooling, containment sprays, residual heat removal, etc. MAAP4 allows operator interventions and incorporates these in a flexible manner, permitting the user to model the operator response and the availability of the various plant systems in a general way.

The user can represent operator actions by specifying a set of values for variables used in the code and/or events, which are the operator intervention conditions. There is a large set of actions that the operator can take in response to the intervention conditions.

MAAP4 has been developed under the FAI Quality Assurance Program, in conformance with 10CFR50 Appendix B and with the International ISO 9000 Standard. Furthermore, the new software has been subjected to review by a Design Review Committee, comprised of senior members of the nuclear community, in a manner similar to that exercised for MAAP 3B.

MAAP4 has been benchmarked against plant experience and large-scale integral experiments and also against one integral computer code. Most of the plant experience and experiment benchmarks are documented in the MAAP4 User's Manual [EPRI, 2003a].

The USNRC reviewed and approved MAAP 3.0B for support of probabilistic risk assessment (PRA) activities at licensed power reactors in the U.S., particularly the IPE's that occurred in the late 1980's and early 1990's. While MAAP4 has not undergone a formal review process by the NRC, the code owner, EPRI, Fauske & Associates, and the MAAP User's Group previously engaged in MAAP4 familiarization activities with the NRC when MAAP4 was first released. Recently, a MAAP4 Information Exchange between these parties has been undertaken in view of the expanding scope of MAAP4 application and MAAP4-supported submittals to the NRC.

MAAP4 has been used previously for safety analyses outside of the risk arena with NRC approval. For example, an NRC Safety Evaluation Report (SER) was written for the D.C. Cook plant in its assessment of minimum safe sump level in the containment recirculation sump during a small LOCA event. This assessment involved small LOCA scenarios that are similar to those in the present analysis for PVNGS.

The MAAP4 RCS model uses momentum equation selectively for sub-models that demand a momentum equation for model integrity. One of the aspects for which a full-fledged momentum equation is not implemented is water flow. Consequently, MAAP4 cannot void the core by reversing flow from the core to the downcomer and loop piping during a large LOCA event. However, small breaks of the size being analyzed for this analysis do not engage in such significant flow reversal, so this limitation is not relevant to this analysis.

The MAAP4 containment model can accommodate most physical phenomena that would occur. However, since it does not entrain pre-existing liquid and condensate from heat sink surfaces, it does not mechanistically bring suspended water droplets into the containment atmosphere (although the model could accommodate droplets if such liquid entrainment was added). Consequently, it conservatively predicts excess gas-phase superheat and pressurization during the blowdown stage of a large LOCA event. Since small breaks of the size being analyzed for this analysis do not engage in this phenomenon, this limitation is not relevant to this analysis. Documented containment benchmarks are testament to the adequacy of the containment model for predicting short-term and long-term containment pressurization under small and medium LOCA conditions, which is necessary for an accurate depiction of containment spray actuation signal (CSAS) timing in this analysis.

The latest MAAP4 archived revision, MAAP 4.0.5 [EPRI, 2003b], was used with the latest PVNGS-specific plant model (a.k.a., parameter file). [

]

[ ]

[ ]

[ ]

The analyses provide three key results. The first result is the RCS pressure that would exist at the time of RAS for various size breaks. These results are provided in Figure 5-1.

Break Size	RCS Pressure at RAS (psia) Discharge Leg Breaks	RCS Pressure at RAS (psia) Suction Leg Breaks
1"	1386	1384
2"	546	438
3"	222	233
4"	213	155
5"	132	148
6"	102	79
7"	77	74
8"	47	53
9"	49	46
10"	37	38

Table 5-1 RCS Pressure at RAS for Various Break Sizes from CENTS

This parameter is used in the following section to [ ] assess ECCS performance (i.e. HPSI flow) under the maximum predicted air ingestion conditions.

The second result from these analyses is that break sizes of 2" diameter or smaller [ ] alternate method of core cooling is available should the HPSI pump fail due to air ingestion. The current PVNGS Emergency Operating Procedures fully implement this recovery strategy.

[ ]

5.2 Determination of Degraded HPSI Flow

[ ]

[ ]

The resulting HPSI system performance or operating points, given the degraded pump performance and the system resistance curves developed above, can be determined and illustrated graphically as shown in Figure 5-2. The developed head and flow rate of the degraded pump is determined by the intersection of the system curves and the degraded pump curves.

[

]

As indicated in Figure 5-2, the static head associated with the 1" diameter small break LOCA at the time of RAS is well above the developed head of the degraded HPSI pump under maximum air ingestion.

For break sizes 2" diameter and larger, Figure 5-2 indicates the degraded HPSI pump has sufficient developed head to continue delivering ECCS flow to the RCS for the short time until the volume of air originally resident in the voided piping is exhausted. After the total air volume is ingested, the Phase 3 pump performance tests demonstrated the HPSI pump would recover and return to its normal non-degraded performance. [

]

5.3 HPSI Pump (Emergency Core Cooling) Safety Function Impact Conclusion

From the Phase 3 pump performance tests under air ingestion, a bounding degraded HPSI pump performance curve was developed. The bounding degraded performance curve envelopes the maximum predicted air volume fractions ingested by the HPSI pump, based on Phase 2 scale-model testing. This study then compared the resulting degraded pump performance with the calculated system resistance that would exist at the time of RAS, for the spectrum of break sizes. The comparison indicates the degraded HPSI pump would develop sufficient discharge head to maintain flow to the RCS for all break sizes except for the smallest breaks less than 2". The degraded flow rate delivered to the RCS would only exist [ ] until the air inventory available to be ingested is exhausted, at which time pump performance can be assumed to return to normal. The analyses performed using the CENTS and MAAP codes determined that for the full spectrum of

[ ]

[ ]

Tests were conducted on the representative CS pump by injecting air at rates up to approximately [ ] air volume fraction. This air volume fraction conservatively bounds the amount of air predicted by scale model testing for all scenarios tested. The pump experienced a reduction in flow during the period of air ingestion, and then returned to normal baseline performance after air injection was suspended. It is concluded that the voided pipe condition does not have a significant impact on Containment Spray pump functionality.

6 Other Considerations

6.1 Waterhammer

The ECCS voided piping condition did not present any negative impacts stemming from waterhammer. Numerous analyses and experiments (References 12 through 14) have been performed to evaluate the influence of air in a system during a strong hydraulic transient such as a pump start. As stated by Martin (Ref. 12):

The effect of the presence of entrapped air on transient pressures of a liquid pipeline can either be beneficial or detrimental, depending on the amount of air, the two-phase flow regime of the mixture (whether homogeneous or slug), and the nature and cause of the transient.

Of particular importance are those situations which could be detrimental to the piping system. Generally these are conditions in which a significant coherent gas volume has formed on the discharge side of the pump. Significant means a volume that is comparable to or larger than the integrated volumetric flow discharged from the pump during the time that it comes up to speed. Given these conditions the pump can accelerate to essentially runout flow conditions with the only resistance being the frictional forces generated by the moving water column between the pump discharge and the air pocket. Subsequent to this, the moving water column will begin to compress the air volume and the gas pressure will increase dramatically as volume is reduced.

For example, under these conditions, the gas bubble pressure more than doubles when the gas volume is reduced by one half and similarly more than doubles again when it is reduced again by one half, etc. Hence, with a low pressure gas volume on the discharge side of the pump, the compression of the gas bubble will eventually absorb the kinetic energy of the water column. For this to occur, the gas volume pressure can increase to values much greater than the maximum pump discharge pressure.

Conversely, if the air volume is on the suction side of the pump such as in the case of the Palo Verde ECCS voided piping, [

]

6.2 Net Positive Suction Head

NUREG/CR-2792 (Ref. 1) provides discussion and guidance regarding the affect of pump air ingestion on NPSH considerations. For example, Section 3.2.3 states that "the presence of air at the inlet.....increases the limiting NPSH required for satisfactory operation. The increased degradation at the pump inlet, as inlet NPSH or pressure is lowered, results from the increased volumetric expansion of air between the pump inlet flange and the impeller inlet. Thus pumps operating with air ingestion will have higher NPSH requirements than those required in single-phase operation."

Section 4.2 goes on to establish an "arbitrary relationship" for the purpose of minimizing this volumetric expansion that occurs between the inlet and the impeller eye. The relationship is:

$$NPSHR_{air/water} = NPSHR_{water} + (1 + 0.5\ AF)$$

Where AF is the air volume fraction in percent. It is noted that this relationship is only intended for use with air volume fractions less than 2%

[redacted]

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[ ]

7 Probabilistic Risk Assessment

Probabilistic Risk Assessment Conclusion

From the CENTS thermal-hydraulics analyses and the Phase 3 pump performance tests, modifications to the Palo Verde Probabilistic Risk Assessment (PRA) model were made to assess the risk significance of the voided pipe condition. The Palo Verde model contains an event tree for small break LOCAs of 2.3 inch diameter and smaller. The model was revised by inserting a failure of the HPSI pumps at RAS (failing the high pressure recirculation function) for small-break LOCA due to air binding, and modeling the subsequent plant cool down and depressurization and LPSI alignment for low pressure recirculation. Consideration was also given to small LOCA events that are induced through the lifting of a PSV and the subsequent failure to reseat. An estimate of the risk increase due to small LOCAs resulting from seismic events was also calculated. Since the pump performance tests indicate that for breaks 2 inches in diameter and larger failure of the HPSI pump is not likely, medium and large LOCA events were unaffected by the voided condition. Thus the small LOCA event would be the dominant contributor to the risk increase due to the voided pipe condition.

[ ] calculated the increase in risk associated with the unfilled containment sumps suction lines. The following table shows the overall impact of loss of High Pressure Recirculation (HPSR) for break sizes of two inches or less.

Initiator	Delta-CDF (per year)
Small LOCA	4.5E-6
PSV – Internal Events Plus Fire	2.0E-6
Seismic	4.7E-7
Total	**7.0E-6**

Table 7-1 Over-all Risk Associated with Loss of HPSR

The above described model adjustments were applied to the entire range of small break LOCA events (i.e. 2.3 " diameter and smaller). The pump testing and analysis program described in the previous sections of this report demonstrate that continued functionality of the HPSI pump for the upper end of the SBLOCA range (those breaks approaching 2" in diameter and larger) would be expected. For the small end of the SBLOCA range of approximately 0.5" in diameter or less, analyses using the CENTS and MAAP code demonstrate that complete depressurization of the RCS to shutdown cooling conditions would be achieved prior to RAS. Therefore, no additional risk is associated with the breaks on the small end of the SBLOCA range. Therefore, the above result provided in Table 7-1 is considered to be a conservative estimate of the incremental risk associated with the ECCS voided piping condition.

8 Conclusions

A comprehensive testing and analysis program was conducted to conservatively estimate the risk significance of the ECCS voided piping condition. The scale model testing program simulated bounding conditions and parameters to provide high confidence the air ingestions rates obtained from the tests exceeded the air ingestion rates the ECCS pumps would have actually experienced had an accident requiring containment recirculation actually occurred. Subsequent pump performance tests were conducted under conditions considered to be more severe than would have been experienced during an actual emergency. The results of the pump performance tests were then used in a set of thermal hydraulic analyses of the Palo Verde Reactor Coolant System and Containment. The analyses determined that performance of the ECCS and containment and temperature control functions would have been maintained. For most postulated accidents scenarios, the ECCS safety function would have been maintained by the HPSI pumps. For a subset of SBLOCA scenarios, the ECCS function would have been maintained by the use of any available CS or LPSI pump following RCS cooldown and depressurization by the Plant Operators, if the HPSI pumps were to have failed due to air ingestion. Utilizing the results of the testing and analysis program in a conservative manner, the incremental risk associated with the ECCS voided piping condition is estimated to be 7.0×10^{-6}.

9 References

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[ ]

ATTACHMENT 2-A

FAI/04-65, Revision 0
Test Report for Phase 1 of Experimental Investigation of Post RAS Air Intrusion into ECCS Suction Piping for Palo Verde Nuclear Generating Station
~~**(Proprietary)**~~
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FAUSKE & ASSOCIATES, INC.

CALCULATION NOTE COVER SHEET

SECTION TO BE COMPLETED BY AUTHOR(S):

Calc-Note Number FAI/04-65 Revision Number 0

Title Test Report for Phase 1 of Experimental Investigation of Post RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station

Project Arizona Public Service (APS) Project Number or Shop Order APS003

Purpose: This report documents the scaled experiments that were conducted to investigate a past plant operability question regarding the possibility of any of the air initially residing in the horizontal segment of the sump suction line being swept into the vertical downcomer and subsequently into the ECCS pumps. The nature of the two phase flow pattern produced in the vertical segment was also investigated.

Results Summary: A range of containment overpressure and system flow rates were investigated. The set of conditions that would be expected for a large break LOCA event were found to result in the air being relocated from the horizontal segment into the vertical segment. The two-phase flow pattern in the vertical segment was seen to be liquid continuous with dispersed air bubbles.

References of Resulting reports, Letters, or Memoranda (Optional)

Author(s): Name (Print or Type)	Signature	Completion Date
R. J. Hammersley	Robert J. Hammersley	September 17, 2004

SECTION TO BE COMPLETED BY VERIFIER(S):

Verifier(s): Name (Print or Type)	Signature	Completion Date
W. E. Berger	[signature]	September 24, 2004

Method of Verification: Design Review ____, Independent Review or Alternate Calculations X, Testing ____

Other (specify) ____

SECTION TO BE COMPLETED BY MANAGER:

Responsible Manager: Name (Print or Type)	Signature	Approval Date
R. E. Henry	Robert E. Henry	September 24, 2004

CALC NOTE NUMBER FAI/04-65 PAGE 2

CALCULATION NOTE METHODOLOGY CHECKLIST

CHECKLIST TO BE COMPLETED BY AUTHOR(S) (CIRCLE APPROPRIATE RESPONSE)

1. Is the subject and/or the purpose of the design analysis clearly stated?... YES (circled) NO
2. Are the required inputs and their sources provided? ... YES (circled) NO N/A
3. Are the assumptions clearly identified and justified? ... YES (circled) NO N/A
4. Are the methods and units clearly identified? ... YES (circled) NO N/A
5. Have the limits of applicability been identified? ... YES (circled) NO N/A
   (Is the analysis for a 3 or 4 loop plant or for a single application.)
6. Are the results of literature searches, if conducted, or other background data provided? ... YES (circled) NO N/A
7. Are all the pages sequentially numbered and identified by the calculation note number?.. ... YES (circled) NO
8. Is the project or shop order clearly identified? ... YES (circled) NO
9. Has the required computer calculation information been provided? ... YES NO N/A (circled)
10. Were the computer codes used under configuration control?... YES NO N/A (circled)
11. Was the computer code(s) used applicable for modeling the physical and/or computational problems identified? ... YES NO N/A (circled)
    (i.e., Is the correct computer code being used for the intended purpose.)
12. Are the results and conclusions clearly stated?... YES (circled) NO
13. Are Open Items properly identified ... YES NO N/A (circled)
14. Were approved Design Control practices followed without exception? ... YES NO N/A (circled)
    (Approved Design Control practices refers to guidance documents within Nuclear Services that state how the work is to be performed, such as how to perform a LOCA analysis.)
15. Have all related contract requirements been met? ... YES (circled) NO N/A

NOTE: If NO to any of the above, Page Number containing justification ______

FAI/04-65

Test Report for Phase 1 of Experimental Investigation of Post RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station

Prepared For:

Palo Verde Nuclear Generating Station

Arizona Public Service

Prepared By:

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September, 2004

ABSTRACT

This report documents the scaled experiments that were conducted to investigate a past plant operability question regarding the possibility of any of the air initially residing in the horizontal segment of the sump suction line being swept into the vertical downcomer and subsequently into the ECCS pumps. The nature of the two phase flow pattern produced in the vertical segment was also investigated.

A range of containment overpressure and system flow rates were investigated. The set of conditions that would be expected for a large break LOCA event were found to result in the air being relocated from the horizontal segment into the vertical segment. The two-phase flow pattern in the vertical segment was seen to be liquid continuous with dispersed air bubbles.

PURPOSE

The purpose of this report is to document the Phase 1 test conditions and results for the APS experimental investigation of the post RAS air intrusion into ECCS suction piping.

INPUT DATA AND ASSUMPTIONS

The Phase 1 experiments were configured and conducted per the approved test plan (FAI, 2004). The initial conditions, major components, and key dimensions for these tests are described in the test plan.

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1.0 PHASE 1 TEST OBJECTIVES

1.1 Technical Issue

The Palo Verde Nuclear Generating Station (PVNGS) has identified a concern that their sump recirculation flow paths to the Emergency Core Cooling System (ECCS) pumps contain a pocket of air trapped between the sump isolation Motor Operated Valves (MOVs) and check valve that could potentially be forced into the operating pump suction upon an initiation of a Recirculation Actuation Signal (RAS) during a design basis event. PVNGS has requested analysis of this concern to determine:

(1) If any volume of air between the inboard sump isolation valve and the downstream check valve could be forced into the suction of the operating High Pressure Safety Injection (HPSI) and Containment Spray (CS) pumps upon full opening of the sump isolation valves at the time of RAS.

(2) The impact on pump performance if any amount of air from the sump suction piping is injected into the operating pumps.

1.2 Experimental Objectives

An experimental investigation has been initiated to address this technical issue and investigate the two-phase flow patterns for the scaled horizontal and downward vertical flow segments. The objective of the Phase 1 testing was to investigate the potential for the air initially resident in the horizontal sump suction line to be forced into the vertical downward piping section.
[

]

[ ] The Phase 1 tests were configured and performed in accordance to the approved test plan (FAI, 2004).

2.0 PHASE 1 TEST FACILITY

2.1 Physical Arrangement

The test facility that was used for the Phase I testing was composed of [

], a centrifugal pump, piping, valves, and associated instrumentation as indicated in Figure 1. The piping and valves used to establish and visualize the flow pattern development from the initial location between the valves and into the downcomer piping were all 4 inch in diameter. The horizontal segment [

]

The vertical [

]

[

]

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[ ]

Figure 1 Phase 1 test configuration for post RAS air intrusion.

[ ]

2.2 Instrumentation

The test instrumentation is indicated in Figure 1 and listed in Table 1. A personal computer (PC) [ ] was used to collect data during [ ]. Each data channel was sampled at a rate of [ ] The data that was recorded for each test included:

[ ]

[ ] General observations [ ] were made and noted by the test engineers. These observations were used to characterize the [ ]

Table 1

[ ]

Following the first four tests in the test matrix the test data was reduced and plotted. The results were inspected for internal consistency as well as confirmation of the proper functioning of the instrumentation. The data collected on instrument P4 appeared to be contaminated with excessive noise. [ ] Thus, in addition to relocating the P4 pressure transducer it was reoriented such that instead of being at the [ ]

2.3 Scaling Considerations

[ ]

[redacted]

[ ]

3.0 *PHASE 1 TEST MATRIX AND TESTING OBSERVATIONS*

3.1 Initial Conditions and Test Matrix

The initial conditions were as follows:

[ ]

* Relative to the elevation of the center line of the horizontal segment of the pump suction line.

The test matrix as provided in the approved Test Plan was modified based on observations during the Phase 1 tests by the Westinghouse project team and the APS representatives [ ] who were observing the tests. The revised test matrix executed in the Phase 1 testing is provided in Table 2. The key observations for each test included [ ]

3.2 Observations During Phase 1 Testing

During the execution of the Phase 1 test matrix several general observations were made in addition to the key object [ ]

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[ ]

[ ]

4.0 *PHASE 1 TEST RESULTS*

4.1 Key Observations

The key observations for the Phase 1 air intrusion test relate to the specific test objectives. The objectives are to observe the behavior of the air in the initially voided horizontal segment and the nature of the flow pattern produced in the vertical downcomer segment. The observations for the 12 tests performed in the Phase 1 testing regarding these objectives are as follows:

- the air initially resident in the voided horizontal segment is removed from the horizontal segment during the initial transient phase,

- the two-phase flow pattern produced in the vertical segment is found to be liquid continuous with the air dispersed as a bubbly flow.

4.2 Discussion of Results

The test data [ ] for each of the twelve Phase 1 tests were reviewed. Table 3 summarizes the results of this review. Table 3 includes [

]

Table 3
Phase 1 Test Results and Observations

[ ]

[ ]

Figure 3A: Total flow rate (Tests 1-4).

[ ]

Figure 3B: Total flow rate (Tests 5-8).

[ ]

Figure 3C: Total flow rate (Tests 9-12).

Table 4

[ ]

5.0 CONCLUSIONS

The Phase 1 tests results lead to the conclusion that the air void initially contained in the horizontal sump suction piping can be swept down and through the vertical piping in the suction line.

6.0 REFERENCES

FAI, 2004, FAI/04-61, "Test Plan for Experimental Investigation of Post RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station," September.

[ ]

ATTACHMENT 2-B

FAI/04-86, Revision 0
Test Report for Phase 2 of Experimental Investigation of Post RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station
~~(Proprietary)~~
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FAUSKE & ASSOCIATES, INC.

CALCULATION NOTE COVER SHEET

SECTION TO BE COMPLETED BY AUTHOR(S):

Calc-Note Number FAI/04-86 Revision Number 0

Title Test Report for Phase 2 of Experimental Investigation of Post-RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station

Project Arizona Public Service (APS) Project Number or Shop Order APS005

Purpose: This report documents the scaled integral experiments (Phase 2) that were conducted to investigate a past plant operability question regarding the possibility of air initially residing in the horizontal segment of the sump suction line being swept into the pump suction header and ECCS pumps. The nature of the two phase flow patterns in the ECCS suction piping was also investigated.

Results Summary:

[ ]

References of Resulting reports, Letters, or Memoranda (Optional)

Author(s): Name (Print or Type)	Signature	Completion Date
Robert J. Hammersley		February 11, 2005
Robert E. Henry		February 11, 2005

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William E. Berger		February 11, 2005

Method of Verification: Design Review _____, Independent Review or Alternate Calculations X, Testing _____

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Responsible Manager: Name (Print or Type)	Signature	Approval Date
R. E. Henry		February 11, 2005

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CALCULATION NOTE METHODOLOGY CHECKLIST

CHECKLIST TO BE COMPLETED BY AUTHOR(S) (CIRCLE APPROPRIATE RESPONSE)

1. Is the subject and/or the purpose of the design analysis clearly stated?... (YES) NO

2. Are the required inputs and their sources provided? ... (YES) NO N/A

3. Are the assumptions clearly identified and justified? ... (YES) NO N/A

4. Are the methods and units clearly identified? ... (YES) NO N/A

5. Have the limits of applicability been identified? ... (YES) NO N/A
(Is the analysis for a 3 or 4 loop plant or for a single application.)

6. Are the results of literature searches, if conducted, or other background data provided? ... (YES) NO N/A

7. Are all the pages sequentially numbered and identified by the calculation note number?... (YES) NO

8. Is the project or shop order clearly identified?... (YES) NO

9. Has the required computer calculation information been provided? ... YES NO (N/A)

10. Were the computer codes used under configuration control?... YES NO (N/A)

11. Was the computer code(s) used applicable for modeling the physical and/or computational problems identified? ... YES NO (N/A)
(i.e., Is the correct computer code being used for the intended purpose.)

12. Are the results and conclusions clearly stated?... (YES) NO

13. Are Open Items properly identified ... YES NO (N/A)

14. Were approved Design Control practices followed without exception? ... YES NO (N/A)
(Approved Design Control practices refers to guidance documents within Nuclear Services that state how the work is to be performed, such as how to perform a LOCA analysis.)

15. Have all related contract requirements been met? ... (YES) NO N/A

NOTE: If NO to any of the above, Page Number containing justification __________

FAI/04-86

Test Report for Phase 2 of Experimental Investigation of Post-RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station

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November, 2004

ABSTRACT

This report documents the Phase 2 scaled experiments that were conducted to investigate a past operability question for the Palo Verde plants regarding the possibility of the air initially residing in the horizontal segment of the sump suction line being swept into the vertical downcomer and subsequently into the ECCS and Containment Spray (CS) pumps. The Phase 1 tests (FAI, 2004a) addressed [ ] The nature of the two phase flow pattern produced [ ] for the High Pressure Safety Injection (HPSI), Low Pressure Safety Injection (LPSI), and CS systems was investigated in these Phase 2 tests.

[

]

Test cases were also included with the HPSI and CS pumps running at the time of RAS with the Low Pressure Safety Injection (LPSI) started later. In general these tests demonstrated that most of the air was pulled through the HPSI suction line before the LPSI pump was started. For most of these tests the HPSI pump was assumed to fail and was shutdown when the flow decreased to one-half of the initial value. Some tests were performed to address the possible operator action of keeping the CS pump on one train and shutting down the CS pump on the other train in favor of the LPSI pump if HPSI were to fail on both trains. With this event sequence, stopping the CS pump enabled the air in the lower header to rise up through the downcomer, pass backward through the check valve and be discharged into the sump thus eventually rising to the containment atmosphere. Consequently, there was no air in the header when the LPSI pump was started.

PURPOSE

This report documents the scaled integral experiments (Phase 2) that were conducted to investigate a past operability question regarding the possibility of air initially residing in the horizontal segment of the sump suction line being swept into the pump suction header and ECCS pumps. The nature of the two phase flow patterns in the ECCS suction piping was also investigated.

INPUT DATA AND ASSUMPTIONS

The Phase 2 experiments were configured and conducted per the approved test plan (FAI, 2004b). The initial conditions, major components, and key dimensions for these tests are described in the test plan.

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1.0 PHASE 2 TEST OBJECTIVES

1.1 Technical Issue

The Palo Verde Nuclear Generating Station (PVNGS) has identified a concern. Specifically, all three units have sump recirculation flow paths to the Emergency Core Cooling System (ECCS) pumps which contain a pocket of air trapped between the sump isolation Motor Operated Valves (MOVs) (butterfly valves) and check valve that could potentially be forced into the operating pump suction upon an initiation of a Recirculation Actuation Signal (RAS) during a design basis event. PVNGS has requested analysis of this concern to determine:

(1) If any air volume between the inboard sump isolation valve and the downstream check valve could be forced into the suction of the operating High Pressure Safety Injection (HPSI) and Containment Spray (CS) pumps upon opening of the sump isolation valves at the time of RAS.

(2) The impact on pump performance if any amount of air from the sump suction piping is injected into the operating pumps.

1.2 Experimental Objectives

Phase 1 testing (FAI, 2004a) demonstrated that the flow demand on the containment sump pump suction line following RAS was sufficient [ ] Therefore, Phase 2 experimental investigation was initiated at FAI to investigate the two-phase flow patterns [ ] The objectives of the Phase 2 testing were to investigate the extent of air transport to the HPSI and CS pumps as well as the LPSI pump for those accident sequences where this could be started. Full scaling testing of the pump performance for the resulting air intrusion will be performed in a Phase 3 test facility at Wyle Laboratories in Huntsville, Alabama. In the Phase 2 testing, the [ ] will be observed including [

] The Phase 2 tests were configured and performed in accordance with the approved test plan (FAI, 2004b).

2.0 PHASE 2 TEST FACILITY

2.1 Physical Arrangement

2.1.1 Configuration 2A

[

]

The use of 4 inch diameter (Schedule 40) pipe to represent the 24 inch diameter (Schedule 20 and 30) pipe in the plant defined a linear scale ratio of approximately 1/6 (FAI, 2004b). Thus, the balance of the suction line pipe lengths and valve locations also used a 1/6th scale unless there were other considerations [

]

[redacted]

Figure 1: Phase 2 Test Configuration 2A for Post-RAS Air Intrusion.

[

] Both the HPSI and CS pumps are single stage centrifugal pumps in the test apparatus. For the plants, the HPSI umps are eight stage centrifugal designs.

Table 1

<u>Test Dimensions</u>

[

]

[ ]

2.1.2 Configuration 2B

[ ]

Figure 2: [ ].

[ ]

[ ]

Figure 3: Phase 2 Test Configuration 2B for Post-RAS Air Intrusion.

[ ]

2.1.3 Configuration 2C

[ ]

[ ]

Figure 4: Phase 2 Test Configuration 2C for Post-RAS Air Intrusion.

[ ]

[ ]

[ ]

[ ]

2.2 Instrumentation

The test instrumentation is similar for all three test configurations and is indicated in Figures 1, 2 and 3 and listed in Table 2. A personal computer (PC) [ ] was used to collect data during the transient following the opening of the isolation valves as well as the subsequent steady state recirculation flow that followed. Each data channel was sampled at a rate of once per [ ] which is much faster than the hydraulic transient which takes tens of seconds.]

] Each experiment had the following data recorded:

[ ]

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Digital movie cameras were used to record the flow patterns in the clear piping sections. General observations in the clear pipe sections were made and noted by the test engineers. These observations were particularly important to characterize the water-air flow patterns in the various suction pipes.

2.3 Scaling Considerations

The test plan (FAI, 2004b) presented the scaling assessment for the Phase 2 tests. The scaling assessment addressed [

] The scaling considerations are discussed below.

2.3.1 Two-Phase Flow Pattern Considerations

[

]

[ ]

2.3.2 [ ]

[ ]

[ ]

[redacted]

[redacted]

[ ]

2.3.3 [ ]

[ ]

[redacted]

[ ]

[redacted]

2.3.4 Materials

[redacted]

[ ]

3.0 *PHASE 2 INITIAL CONDITIONS AND TEST MATRIX*

The range of initial conditions were as follows:

[ ]

The test matrices for Configurations 2A, 2B and 2C as provided in the Phase 2 Test Plan (FAI, 2004b) are reproduced in Tables 3, 4 and 4 respectively. With the observations of the initial tests, the test matrix was expanded during the testing program to investigate specific phenomena as well as demonstrate reproducibility of the results. The expanded test matrix executed in the Phase 2 testing is presented in Section 4.0, **Phase 2 Tests Results**. A cross reference is provided between the expanded test matrix and the test matrix from the test plan. Key observations for each test include the two-phase flow pattern [ ] Other observations include [ ]

* Relative to elevation of center line of the lower horizontal header for the HPSI, CS and LPSI pump suction lines.

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[ ]

4.0 PHASE 2 TEST RESULTS

The test data and digital movies of the Phase 2, Configurations 2A were reviewed for the tests specified for this configuration in the Phase 2 Test Plan (FAI, 2004b). After reviewing and discussing the results from the original twelve tests for Configuration 2A with APS and Westinghouse personnel, it was decided to expand the Configuration 2A test matrix to 29 tests. Table 6 summarizes the results for all 29 of the Configuration 2A tests [

]

[ ]

Upon the completion of the expanded set of Configuration 2A tests, review of the experimental data and the [ ] as well as other supporting plant analyses (Phase 4 of the overall program), it was decided to investigate two other pump combinations. The results for the Configuration 2B and 2C experiments are summarized in Tables 7 and 8. Observations and insights gained from these configurations are discussed after those resulting from Configuration 2A.

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[ ]

4.1 Configuration 2A

4.1.1 Key Observations

The key observations for the Phase 2 air intrusion tests relate to the specific test objectives, i.e. (1) to investigate the air delivery rates to the HPSI and CS pump suctions and (2) document the associated two-phase flow patterns. Observations from the 29 tests performed in Configuration 2A of the Phase 2 testing are as follows:

[ ]

4.1.2 Discussion of Results

4.1.2.1 General Comments

[ ]

[ ]

4.1.2.2 Flow Patterns

[ ]

[ ]

[ ]

[[redacted]]

[redacted]

[redacted]

[redacted]

[ ]

[ ]

[ ]

[redacted]

[redacted]

[ ]

[redacted]

[redacted]

[redacted]

[[redacted]]

[ ]

[redacted]

[redacted]

4.1.2.3.1 Interpretation of the HPSI Air Intrusion Rate

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[ ]

[redacted]

[ ]

[ ]

[redacted]

[redacted]

[ ]

[redacted]

[ ]

4.1.2.3.2 Interpretation of the CS Air Intrusion Rate

[ ]

[ ]

4.1.2.4 [ ]

[ ]

[ ]

4.1.2.5 [ ]

[ ]

[ ]

4.2 Configuration 2B

4.2.1 Key Observations

[ ]

4.2.2 Discussion of Results

4.2.2.1 General Comments

These scoping tests included 8 experiments with the principal difference being the predetermined LPSI flow rate. Table 7 summarizes the results for all tests including the as-tested flow rates for both pumps and the corresponding Froude numbers in the different piping segments. [ ]

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4.2.2.2 Flow Patterns

[redacted]

[ ]

4.2.2.3 Interpretation of the Air Intrusion Rate

[ ]

4.3 Configuration 2C

4.3.1 Key Observations

[ ]

[redacted]

[ ]

[ ]

5.0 CONCLUSIONS

The following conclusions were derived from the three Phase 2 configurations for the integral 4 inch diameter scaled experiments representing the Palo Verde sump suction line behavior.

I. Configuration 2A

[ ]

[ ]

[ ]

II. Configuration 2B

[ ]

III. Configuration 2C

[ ]

[ ]

6.0 REFERENCES

FAI, 2004a, "FAI/04-65, "Test Report for Phase 1 of Experimental Investigation of Post RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generating Station," September.

FAI, 2004b, FAI/04-74, "Test Plan (Phase 2) for Experimental Investigation of Post-RAS Air Intrusion Into ECCS Suction Piping for Palo Verde Nuclear Generation Station," December.

FAI, 2005, FAI/04-79, "Test Report: Phenomenological Studies for the APS Containment Suction Line," January.

ATTACHMENT 2-C

Test Report 10530R01
Test Report for Testing of a CA Pump and WDF Pump with a Void Fraction Inlet Fluid Condition
~~(Proprietary)~~

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TEST REPORT

REPORT NO. 10530R01, Revision 0
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DATE 01/18/05
SPECIFICATION(S) WLTP 10530TP, Revision 0

TEST REPORT FOR TESTING OF A CA PUMP AND A WDF PUMP WITH A VOID FRACTION INLET FLUID CONDITION

Arizona Public Service (APS)

STATE OF ALABAMA } SS. Alabama Professional Eng
COUNTY OF MADISON Reg. No. 25471

Ralph D. Yeardley, being duly sworn, deposes and says: The information contained in the report is the result of complete and carefully conducted tests and is to the best of his knowledge true and correct in all respects.

[signature]

SUBSCRIBED and sworn to before me this 21 day of Jan, 2005

Ardella Stebbins SEAL
Notary Public in and for the State of Alabama at large

My Commission expires March 3, 2007

Wyle shall have no liability for damages of any kind to person or property, including special or consequential damages, resulting from Wyle's providing the services covered by this report.

PREPARED BY G. Smyth 1/21/05 Date

APPROVED BY [signature] 1/21/05 Date

WYLE Q. A. [signature] 2/8/05 Date

wyle laboratories

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Attachments

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1.0 CUSTOMER Arizona Public Service Co.

ADDRESS Palo Verde Nuclear Generating Station
Tonpah, AR 85354.

2.0 TEST SPECIMEN The equipment to be tested consists of two ESF (Emergency Safety Feature) pump / motor assemblies; the CA pump and motor and the WDF pump and motor.

3.0 MANUFACTURER The pumps were manufactured by Ingersoll-Rand. The motors were manufactured by Westinghouse.

4.0 SUMMARY

This document has been prepared by Wyle Laboratories to document the results of a test program on the CA and WDF pumps and motors to determine the performance with a void fraction inlet fluid condition.

This testing was performed in accordance with Wyle Laboratories Test Procedure 10530TP, "Test Procedure for Testing of a CA pump and a WDF pump with a Void Fraction Inlet Fluid Condition". The testing meets the requirements of the APS Purchase Order 500281122.

5.0 EQUIPMENT DESCRIPTION

The equipment to be tested consists of two pump / motor assemblies; the CA pump and motor, and the WDF pump and motor.

Description: The equipment description is as follows:

CA

Motor (CA):
Westinghouse Electric Frame 5810H
Class 1E
Rated at 1000 HP, 3-Phase, 60 Hz, 4000 Volts
Speed: 3553 rpm
Weight: 4,800 lbs
Motor Identification Number: 17535LN01

Pump (CA):
4x11CA-8
Nameplate Head = 2850 ft
Horizontal shaft
Nameplate Rated flow = 900 gpm
Weight: 4,400 lbs
Suction diameter: 10" sch 40
Discharge diameter 4" sch 80

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Attachment A1

Receiving Inspection

RESULTS

Receiving inspections were performed on November 22, 2004 for both the CA and WDF motor and pump assemblies upon receipt at Wyle Laboratories in accordance with section 3.1 of Wyle Laboratories Test Procedure No. 10530, Revision 0.

The CA pump and motor arrived as two individual pieces. The coupling and miscellaneous spare parts were supplied with the pump.

The WDF pump and motor arrived as three pieces; the inlet piping and pump casing assembly, the motor assembly and a box of miscellaneous parts including the seal piping and impeller.

The nameplate data and results of the inspection were recorded on the attached Receiving Inspection Data Sheet.

The specimen pump and motor assemblies were as described in paragraph 5.0 of this report.

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RECEIVING INSPECTION DATA SHEET

PUMP DATA

TAG NO.:	087634	SERIAL NO:	087634
MANUFACTURER:	Ingersoll-Rand	RATED FLOW:	[ ]
NOMINAL SIZE:	8 x 20 WDF	SHUT OFF HEAD:	[ ]
END CONNECTION:	[ ]		
	[ ]		
	[ ]		

MOTOR DATA

MANUFACTURER:	Westinghouse	FRAME:	55010-P39
MODEL #: (ID)	VSWF	SERIAL #:	1S-78
INS. CLASS:	B	VOLTAGE:	4000
CURRENT @RATED VOLTAGE	62	SPEED:	1776
FREQUENCY:	60 Hz		

DESCRIBE CONDITION OF RECEIVED ITEM:

Motor received on metal pallet marked #2B. Pump casing on pallet #1B. Received box containing diffuser, pump seal piping, struts, electric box and hardware.

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RECEIVING INSPECTION DATA SHEET

PUMP DATA

TAG NO.:	N/A	SERIAL NO:	117814/547
MANUFACTURER:	Ingersoll-Rand	RATED FLOW:	[ ]
NOMINAL SIZE:	4 x 11 CA x 8	SHUT OFF HEAD:	[ ]
END CONNECTION:	[ ]		
	[ ]		
	[ ]		

MOTOR DATA

MANUFACTURER:	Westinghouse	FRAME:	5810H
MODEL #: (ID)	HSW2	SERIAL #:	17535LN01
INS. CLASS:	F	VOLTAGE:	4000
CURRENT @RATED VOLTAGE	123	SPEED:	3553
FREQUENCY:	60 Hz		

DESCRIBE CONDITION OF RECEIVED ITEM:

Pipe on bottom of pump appears to be bent. Miscellaneous parts with pump include plates and all thread. Seals, gaskets, and spare bearings included. SS shaft is s/n 557. Coupler sleeve and coupler both have tags with 62013784. Motor received on pallet.

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Attachment A2

Test Facility Description, Equipment Setup for testing and Instrumentation

RESULTS

Test Facility Description

[

]

The overall test facility is illustrated in Figure 1 in Attachment A6. [

]

[

]

CA Pump/Motor Equipment Setup for Testing

The CA pump and motor were setup in accordance with section 3.2 of Wyle Laboratories Test Procedure No. 10530TP, Revision 0.

From December 01 to 09, the following activities were completed:

[

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Instrumentation

Following the CA test specimen pump and motor installation and alignment, the instrumentation was installed.

The following table summarizes the instrumentation used for the test program and the identification numbers (TAG) used by Wyle Laboratories:

CA Pump Loop Instrumentation:

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[ ]

All Wyle Laboratories' test equipment is calibrated on a periodic basis with the calibration interval displayed on a decal. This decal is affixed to the equipment indicating the last calibration date, the next calibration due date, accuracy, and by whom calibrated. The instrumentation equipment sheet for all the instrumentation is presented in this attachment.

[ ]

Instrumentation

Following the WDF test specimen pump and motor installation, the instrumentation was installed.

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Key of Attachments:

Instrumentation Sheet for Test Program. (3 pages)

Calibration Data for the Turbine Flow Meters. (2 pages)

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Attachment A3

Checkout Testing

RESULTS

Prior to the actual testing, a test facility and test specimen check out was performed to verify facility capabilities, test specimen operation and instrumentation functionality for the two test loops.

During this checkout test program, the data channels were acquired at [ ] sample frequency. A Test Log datasheet was used to record test run descriptions, as well as test data and time information and the [ ] conditions. The test log datasheets obtained during the check out testing are presented in this attachment.

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In all cases, the data files have been supplied to APS separately.

[ ]

An instrumentation equipment sheet for the testing is presented in attachment A2.

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Key of Attachments:

Test Log Sheets for the Check Out Testing (4 pages)

Start up Check list (CA pump test) (2 pages)

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TEST LOG SHEET

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Attachment A4

Performance Testing

RESULTS

The intent of the testing was to determine if temporary performance degradation occurs during the ingestion of a void fraction, and to identify any permanent degradation of performance after un-voided inventory returns to the pump.

A summarized test matrix for both pumps is presented in this Attachment.

During the test program, the data channels described in Attachment A3 were acquired [ ] by the Wyle Laboratories data acquisition system. A Test Log datasheet was used to record test run descriptions, as well as test data and time information and the ambient temperature, pressure and flow conditions. The log is presented in Attachment A3.

[ ]

The instrumentation equipment sheet for this testing is presented in Attachment A2.

[ ]

Actual Test Matrix

Throughout the test program, the required data described in Attachment A2 was recorded. This data covers the complete test program. Note that the test matrix presented here represents the target data for testing. [ ]

The actual test data files consisting of [ ]

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CA and WDF Pump Test Matrix

CA Pump Test Matrix

WDF Pump Test Matrix

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A summary of the actual test data plots is presented here for the following test cases; 1D rerun, 2E, 3C and 4B.

DataSet for Test 1D rerun

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DataSet for Test 2E

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DataSet for Test 3C

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Dataset for Test 4B – the WDF pump

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Attachment A5

Equipment Inspection and Shipment

RESULTS

[ ]

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Attachment A6

Photographs

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Attachment A7

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DATA FILE FORMAT

The recorded data is as follows:

[ ]

OVERALL g RMS DATA FOR VIBRATION DATA FILES LISTED

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ATTACHMENT 2-D

FAI/05-06, Revision 0
Summary Report of MAAP4 LOCA Analysis in Support of Past Operability Assessment of Degraded HPSI Performance During Containment Recirculation at Palo Verde
~~**(Proprietary)**~~
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FAUSKE & ASSOCIATES, LLC
CALCULATION NOTE COVER SHEET

SECTION TO BE COMPLETED BY AUTHOR(S):

Calc-Note Number FAI/05-06 Revision Number 0.

Title Summary Report of MAAP4 LOCA Analysis in Support of Past Operability Assessment of Degraded HPSI Performance During Containment Recirculation at Palo Verde.

Project Arizona Public Service Project Number or Shop Order APS007.

Purpose: See Section 1.0.

Results Summary: See Section 5.0.

References of Resulting reports, Letters, or Memoranda (Optional): N/A.

Author(s): Name (Print or Type)	Signature	Completion Date
Christopher E. Henry	Christopher E. Henry	February 11, 2005.

SECTION TO BE COMPLETED BY VERIFIER(S):

Verifier(s): Name (Print or Type)	Signature	Completion Date
G. Thomas Elicson	G. Thomas Elicson	February 11, 2005.

Method of Verification: Design Review ____, Independent Review or Alternate Calculations X, Testing ____.
Other (specify) ____.

SECTION TO BE COMPLETED BY MANAGER:

Responsible Manager: Name (Print or Type)	Signature	Approval Date
Christopher E. Henry	Christopher E. Henry	February 11, 2005.

CALC NOTE NUMBER FAI/05-06 PAGE ii

CALCULATION NOTE METHODOLOGY CHECKLIST

CHECKLIST TO BE COMPLETED BY AUTHOR(S) (CIRCLE APPROPRIATE RESPONSE)

1.	Is the subject and/or the purpose of the design analysis clearly stated?	(YES)	NO	
2.	Are the required inputs and their sources provided?	(YES)	NO	N/A
3.	Are the assumptions clearly identified and justified?	(YES)	NO	N/A
4.	Are the methods and units clearly identified?	(YES)	NO	N/A
5.	Have the limits of applicability been identified? (Is the analysis for a 3 or 4 loop plant or for a single application.)	(YES)	NO	N/A
6.	Are the results of literature searches, if conducted, or other background data provided?	(YES)	NO	N/A
7.	Are all the pages sequentially numbered and identified by the calculation note number?	(YES)	NO	
8.	Is the project or shop order clearly identified?	(YES)	NO	
9.	Has the required computer calculation information been provided?	(YES)	NO	N/A
10.	Were the computer codes used under configuration control?	(YES)	NO	N/A
11.	Was the computer code(s) used applicable for modeling the physical and/or computational problems identified? (i.e., Is the correct computer code being used for the intended purpose.)	(YES)	NO	N/A
12.	Are the results and conclusions clearly stated?	(YES)	NO	
13.	Are Open Items properly identified	YES	NO	(N/A)
14.	Were approved Design Control practices followed without exception? (Approved Design Control practices refers to guidance documents within Nuclear Services that state how the work is to be performed, such as how to perform a LOCA analysis.)	YES	NO	(N/A)
15.	Have all related contract requirements been met?	(YES)	NO	N/A

NOTE: If **NO** to any of the above, Page Number containing justification

FAI/05-06

SUMMARY REPORT OF MAAP4 LOCA ANALYSIS IN SUPPORT OF PAST OPERABILITY ASSESSMENT OF DEGRADED HPSI PERFORMANCE DURING CONTAINMENT RECIRCULATION AT PALO VERDE

Submitted To:

Arizona Public Service
Phoenix, Arizona

Prepared By:

Christopher E. Henry
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February 2005

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[redacted]

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EXECUTIVE SUMMARY

This report documents MAAP4 calculations of Palo Verde Nuclear Generating Station (PVNGS) core, reactor coolant system (RCS), and containment thermal-hydraulic response to a small-to-medium loss of coolant accident (LOCA) in which the high-pressure safety injection (HPSI) and containment spray system (CSS) become degraded. Potential failure of HPSI is also considered. Degradation and potential failure are presumed to occur when the emergency core cooling system (ECCS) and CSS transition between suction from the refueling water tank (RWT) to suction from the containment recirculation sump in response to the recirculation acquisition signal (RAS). This scenario is intended to support a justification for past operations (JPO) assessment regarding degradation and possible failure of the HPSI system due to ingestion of air that actually existed between two valves in the ECCS/CSS suction lines during past operation of the plant.

Specifically, a spectrum of break sizes and locations was evaluated to determine the case(s) that could challenge core coverage, long-term core cooling, and long-term containment heat removal. The medium break diameters in the range of roughly 3 to 6 inches were determined to be the most challenging. However, in all cases, MAAP4 predicted that the core would remain completely covered, due almost entirely to the cold leg injection of the safety injection tanks (SIT) (a.k.a., accumulators) during the post-RAS time period. Even when outright post-RAS failure of the HPSI was postulated, SIT injection maintained core coverage until post-LOCA cooldown and depressurization of the RCS below the low-pressure safety injection (LPSI) shutoff head enabled sufficient LPSI flow to provide continued core coverage and long-term core cooling.

1.0 INTRODUCTION

1.1 Background

On September 28, 2004, PVNGS staff [PVNGS, 2004a] submitted a licensee event report (LER) to the Nuclear Regulatory Commission (NRC) that reported a condition in Units 1, 2, and 3 in which air voids in the recirculation sump suction piping (serving both the ECCS and the CSS) may have prevented the fulfillment of the system safety function to removal residual heat and mitigate the consequences of a loss of coolant accident. (Reference [Westinghouse, 2004] provides some additional details that are relevant to all Westinghouse and CE designs.)

PVNGS, in conjunction with Westinghouse and its Fauske and Associates (FAI) subsidiary, investigated this condition with an approach that involved both experiment and analytical elements. Phases 1 through 3 of the investigation were predominantly experimental separate effects testing of HPSI/CSS availability and are not considered here. Phase 4 was the integral plant analysis with independent evaluations provided by the MAAP4 and CENTS codes. ~~This report is confined to MAAP4 analysis portion of Phase 4.~~

Phase 4 participants from PVNGS, Westinghouse (Windsor, Connecticut office), and FAI were charged with considering the core, RCS, and containment response to post-RAS degradation and potential failure of the HPSI and CSS. Furthermore, this circumstance could result from any of the full spectrum of initiating events (LOCA, transient, station blackout, ...) that would challenge core coverage, long-term core cooling and, long-term containment heat removal (and by extension long-term containment integrity). Since the outcome of challenges could involve core overheat and damage, the MAAP4 code was selected as a contributor to the analysis in view of its ability to model degraded core progression and its influence on the RCS and containment.

1.2 Post-RAS ECCS and CSS Status

It has been established that the HPSI system within the ECCS and the CSS could become degraded or even unavailable during post-RAS operation due to ingestion of pre-existing air within the suction lines. Elaboration on some key details is instructive.

At the time of RAS, the PVNGS units are designed for automatic switchover of the HPSI and CSS systems. Specifically, these systems are stopped, realigned to the recirculation sump, and then restarted during the automatic switchover. The LPSI system is stopped as part of this process, but it is not automatically restarted. It must be manually restarted by the operator (if necessary) after completion of switchover. Furthermore, the HPSI suction line is the first system to draw from the suction header. This is followed by the CSS suction line and finally the LPSI suction line. Also, the specific configuration of the HPSI suction line makes HPSI more susceptible to air ingestion than the other systems.

Indeed, the noted Phase 1 and Phase 2 experiments, which were responsible for characterizing the two-phase flow through the suction header and individual ECCS/CSS suction lines, demonstrated that most air ingestion would occur in the HPSI system with only a relatively small ingestion by the CSS system.

Phase 3 experiments were responsible for evaluating an actual HPSI pump with air ingestion boundary conditions dictated by Phase 1 and Phase 2 experiments. These experiments demonstrated that the HPSI system would continue to operate but at a degraded flow condition, with increasing degradation (decreasing flow) at higher system pressure.

Therefore, Phase 4 analyzed both degraded and failed conditions for HPSI, a prescribed degraded condition for CSS and full availability of LPSI in the post-RAS operation. Specific details will be provided in Section 3.

1.3 Initiating Event Selection

As stated above, all initiating events were considered which would challenge core coverage, long-term core cooling, and long-term containment heat removal. Furthermore, the Level II containment event trees [PVNGS, 1992] for these initiating events were inspected to determine the most challenging set of conditions for high-pressure recirculation degradation or failure. Note, evaluation of the event trees did not entail loss of additional components concurrent with the HPSI degradation or failure. Since this was a deterministic (as opposed to probabilistic) analysis that was intended to support justification for past operation, all other systems were assumed to be available, particularly the safety injection tanks (SIT) and the operator action of post-LOCA steam generator cooldown and depressurization of the RCS via the steam generators.

1.4 Break Size and Location Selection

With these ground rules in place, it was determined that a small to medium LOCA ~~(roughly 3 to 6 inches in diameter) initiating event is most challenging since it is responsible for~~ significant coolant loss, but the RCS remains at elevate pressure because the break alone is not sufficient to remove decay power. [

]

[redacted]

2.0 MAAP CODE DESCRIPTION

2.1 What is MAAP?

MAAP is a computer code that simulates light water reactor system response to accident initiation events. The Modular Accident Analysis Program (MAAP), an integral systems analysis computer code for assessing severe accidents, was initially developed during the industry-sponsored IDCOR Program. At the completion of IDCOR, ownership of MAAP was transferred to EPRI. Subsequently, the code evolved into a major analytical tool (MAAP 3B) for supporting the plant-specific Individual Plant Examinations (IPEs) requested by NRC Generic Letter 88-20. Furthermore, MAAP 3B was used as the basis to model the Ontario Hydro CANDU designs. As the attention of plant-specific analyses was expanded to include accident management evaluations, the scope of MAAP (its design basis) was expanded to include the necessary models for accident management assessments. Through support by the U.S. Department of Energy (DOE), the MAAP4 design basis was further extended to include the Advanced Light Water Reactor (ALWR) designs currently being developed by the reactor vendors. ~~MAAP4 has also been expanded to represent the VVER designs used in Finland and~~ central Europe.

2.2 MAAP History

Table 2-1 summarizes the history of MAAP development in terms of the major code versions and the major advancements represented by each version. Two types of Nuclear Steam Supply Systems (NSSS) are modeled in the MAAP4 code: the Boiling Water Reactor (BWR) and the Pressurized Water Reactor (PWR). In addition, MAAP4 is the first archived code that contains a graphical representation of the reactor and containment response (MAAP4-GRAAPH). MAAP4, like MAAP 3B, is currently being maintained by Fauske & Associates, LLC (FAI) for the Electric Power Research Institute (EPRI) and the MAAP User's Group (MUG).

Table 2-1: History of MAAP Code Development.

Year	MAAP Code Version	Major Advancement
1982	-	MAAP development initiated for BWRs and PWRs.
June, 1983	1.0	Primary system and containment thermal-hydraulic models.
June, 1984	2.0	Fission product release, transport and deposition models added; local H_2 burning (igniters).
December, 1984	2.0B	Zircaloy-tellurium binding.
January, 1986	3.0	In-vessel natural circulation, advanced models for aerosol growth and deposition, suppression pool scrubbing, gas natural circulation in steam generation, Chexal/Layman correlation for BWR core power model.
January, 1988 (MAAP Users' Group Initiated)	3.0B	Auxiliary building/reactor building model, improved suppression pool scrubbing model, increased RCS nodalization, RCS natural circulation.
1991	MAAP-CANDU	CANDU-specific models for the horizontal fuel bundle and pressure tubes, moderator tank, shield tank, multi-unit containment, and vacuum building.
September, 1993	MAAP-VVER	Fuel cans for the PWR core, horizontal steam generator, fuel movement as part of the shutdown mechanism.
May, 1994	MAAP4 MAAP4-GRAAPH MAAP4-DOSE	Accident management and ALWR models, advanced core melt progression and material creep models, in-vessel cooling, external cooling of the RPV, detailed modeling of the lower head penetrations, generalized containment, interactive graphical interface, on-site and off-site radiation dose models.

The purpose of MAAP4 is to provide an accident analysis code that can be used with confidence by the nuclear industry in all phases of severe accident studies, including accident management, for current reactor/containment designs and for ALWRs. MAAP4 includes models for the important accident phenomena that might occur within the primary system, in the containment, and/or in the auxiliary/reactor building. For a specified reactor and containment system, MAAP4 calculates the progression of the postulated accident sequence, including the disposition of the fission products, from a set of initiating events to either a safe, stable state or to an impaired containment condition (by overpressure or over-temperature) and the possible release of fission products to the environment.

Severe accident analyses can be divided into four phases: (1) prevention of core damage; (2) recovery prior to reactor pressure vessel breach; (3) recovery after vessel breach, but prior to containment failure; and (4) mitigation of releases of fission products reaching reactor/auxiliary buildings. The previous archived version, MAAP 3B, can analyze phases 1, 3, and 4 for existing reactors, which is sufficient to support the Individual Plant Examination (IPE) studies, the intended purpose of that major MAAP version. However, MAAP 3B does not have the ability to treat phase 2, recovery prior to vessel breach but after severe core damage. It has been estimated that the interval between the onset of severe core damage and the time of vessel breach could vary from 30 minutes to many hours or, as in the TMI-2 accident, vessel integrity can be maintained throughout the accident. Recovery during this interval could obviously reduce, and perhaps eliminate, the likelihood of reactor pressure vessel failure and thereby greatly limit the extent of the accident.

In evaluating the effectiveness of proposed accident management strategies, there is a need to evaluate the integral system response to the proposed actions. Because of the numerous phenomena involved the evaluation is complex, and for many severe accident phenomena, the experimental database is sparse. However, with the extensive TMI-2 data, along with the results of integral experiments such as the LOFT and CORA tests, the major characteristics of the melt progression, primary system thermal-hydraulic response, and core debris-concrete interaction have been demonstrated. Also, with EPRI-sponsored experiments, more data have become available on key phenomena, for example, the mode of vessel breach and the conditions which could prevent vessel failure. The results from these experiments have been included in the MAAP4 modeling enhancements and have resulted in major insights with respect to the effectiveness of accident management actions, particularly for maintaining the integrity of the reactor vessel.

One area where only limited experimental data are available is quenching of overheated debris prior to vessel breach. This of course, is of key interest in recovering from an accident state and was a major part of the TMI-2 accident. MAAP4 includes models for in-vessel cooling and external cooling of the RPV to evaluate whether a safe, stable state can evolve following water addition to the RCS and/or the containment if the core debris can be retained within the reactor pressure vessel.

MAAP4 also addresses the new and unique features, many of which are passive, included in ALWR designs. These are:

- passive heat removal system, such as an in-containment isolation condenser or a passive RHR system,
- gravity-fed water injection systems,
- external heat removal from the containment shell,
- a generalized nodalization scheme for the containment to accommodate the ALWR designs including an in-containment RWST, and
- the capability to analyze flow through large safety valves, such as an automatic depressurization system for PWR designs.

Since the beginning of the MAAP code development, the codes have represented all of the important safety systems such as emergency core cooling, containment sprays, residual heat removal, etc. MAAP4 allows operator interventions and incorporates these in a flexible manner, permitting the user to model the operator response and the availability of the various plant systems in a general way. The user can represent operator actions by specifying a set of values for variables used in the code and/or events, which are the operator intervention conditions. There is a large set of actions that the operator can take in response to the intervention conditions.

MAAP4 has been developed under the FAI Quality Assurance Program, in conformance with 10CFR50 Appendix B and with the International ISO 9000 Standard. Furthermore, the new software has been subjected to review by a Design Review Committee, comprised of senior members of the nuclear community, in a manner similar to that exercised for MAAP 3B.

2.3 Summary of Relevant Benchmarks

The following subsections provide a summary of relevant MAAP4 benchmarks against plant experience and large-scale integral experiments and also against one integral computer code. Plant experience and experiment benchmarks are documented in Volume 3 of the MAAP4 User's Manual [EPRI, 2003a]. (The MB-2 benchmark is awaiting incorporation into the manual in the next MAAP4 revision cycle this year.)

2.3.1 RCS Response to Small LOCA

Since RCS thermal-hydraulic performance under a small LOCA condition is essential to the analysis, some relevant benchmarks are cited here.

MAAP4 RCS thermal-hydraulics has been benchmarked against the Three Mile Island Unit 2 (TMI-2) plant experience, particularly the small LOCA phase of the accident when the pressurizer relief valve was stuck open. MAAP4 RCS thermal-hydraulics has also benchmarked against a similar stuck open pressurizer relief valve event at Crystal River Unit 3. Both benchmarks show reasonable good agreement with the plant data. While these benchmarks are for RCS hot side LOCA's in the pressurizer, they are still relevant to cold side LOCA's since the LOCA modeling in the MAAP pressurizer model is essentially the same as that used for LOCA modeling in RCS loop piping.

As part of the recent Beaver Valley atmospheric containment conversion project, MAAP4 was benchmarked against the Westinghouse small LOCA code, NOTRUMP.

2.3.2 Containment Response to LOCA

Since containment response is an important aspect of RAS timing, it is important to insure the integrity of the MAAP4 containment model. MAAP4 has been benchmarked against numerous containment experiments, both separate effects tests and large-scale integral effects tests. Herein, the containment was benchmarked as a stand-alone model with break mass and energy rates from the experiment, specified as a boundary condition to the model. This type of stand-alone benchmark can be performed within the normal MAAP4 code framework via the MAAP4 dynamic benchmarking feature, thereby exercising the exact same containment model that is used in conventional MAAP4 applications that exercise the full code.

Two benchmarks of note are the small LOCA experiment E11.2 and the medium LOCA experiment T31.5 performed at the HDR test facility in Germany, which was a reactor-scale containment that contained a decommissioned low-power reactor. MAAP4 compares well to both short-term and long-term containment pressurization in both experiments.

2.3.3 RCS Response to Steam Generator Tube Heat Transfer

Since post-LOCA cooldown and depressurization is an important operator action in this analysis, it is important to insure the integrity of the RCS response to steam generator tube heat transfer.

MAAP4 has been benchmarked the Crystal River Unit 3 plant transient, noted above. Herein, steam generators temporarily boiled dry during the transient prior to receiving auxiliary feedwater. Also, in a similar event, the Davis-Besse Unit 1 plant transient resulted in the steam generators boiling dry for a brief period until auxiliary feedwater could be provided. The MAAP4 RCS model, in particular the primary system average temperature, compares well during both the initial steam generator heat transfer and subsequent primary system heatup in the presence of dry steam generators.

The MAAP4 steam generator model has been compared against an integral steam generator experiment known as the Westinghouse Model Boiler 2 (MB-2). Herein, the steam generator is treated as a stand-alone model with primary system boundary conditions from the experiment provided via user input. Again, like the stand-alone containment benchmark, a stand-alone steam generator benchmark can be performed within the normal MAAP4 code framework via the MAAP4 dynamic benchmarking feature, thereby exercising the exact same steam generator model that is used in conventional MAAP4 applications that exercise the full code. Revision MAAP 4.0.5, which is the code revision used for this analysis, was successfully benchmarked against loss of feedwater tests (both simulated full power and decay power transients) performed at MB-2.

2.4 Regulatory Understanding of MAAP

The U.S. Nuclear Regulatory Commission (NRC) reviewed and approved MAAP 3.0B for support of probabilistic risk assessment (PRA) activities at licensed power reactors in the U.S., particularly the individual plant examinations (IPE's) that occurred in the late 1980's and ~~early 1990's.~~

While MAAP4 has not undergone a formal review process by the NRC, the code owner, the Electric Power Research Institute (EPRI), Fauske and Associates (FAI), and the MAAP User's Group (MUG) previously engaged in MAAP4 familiarization activities with the NRC when MAAP4 was first released. Recently, a MAAP4 Information Exchange between these parties has been undertaken in view of the expanding scope of MAAP4 application and MAAP4-supported submittals to the NRC.

MAAP4 has been used previously for safety analyses outside of the risk arena with NRC approval. For example, an NRC Safety Evaluation Report (SER) was written for the D.C. Cook plant in its assessment of minimum safe sump level in the containment recirculation sump during a small LOCA event. This assessment involved small LOCA scenarios that are similar to those in the present analysis for PVNGS.

2.5 MAAP4 Limitations

2.5.1 MAAP4 RCS Model

The MAAP4 RCS model uses momentum equation selectively for sub-models that demand a momentum equation for model adequacy. One of the aspects for which a full-fledged momentum equation is not implemented is water flow. Consequently, MAAP4 cannot void the core by reversing flow from the core to the downcomer and loop piping during a large LOCA event. However, small breaks of the size being analyzed for this analysis do not engage in such significant flow reversal, so this limitation is not relevant to this analysis.

2.5.2 MAAP4 Containment Model

The MAAP4 containment model can accommodate most physical phenomena that would occur. However, since it does not entrain pre-existing liquid and condensate from heat sink surfaces, it does not mechanistically bring suspended water droplets into the containment atmosphere (although the model could accommodate droplets if such liquid entrainment was added). Consequently, it is conservatively predicts excess gas-phase superheat and pressurization during the blowdown stage of a large LOCA event.

Again, small breaks of the size being analyzed for this analysis do not promote significant gas superheat, so this limitation is not relevant to this analysis. Furthermore, superheat and excess pressurization are conservative for this analysis since they would lead to earlier RAS timing. As noted previously, the HDR T31.5 and E11.2 containment benchmarks are testament to the adequacy of the containment model for predicting short-term and long-term containment pressurization under small and medium LOCA conditions, which is necessary for an accurate depiction of containment spray actuation signal (CSAS) timing in this analysis.

2.6 Refinements to the MAAP4 Code Revision

The latest MAAP4 archived revision, MAAP 4.0.5 [EPRI, 2003b], was used with the latest PVNGS-specific plant model (a.k.a., parameter file). [

]

[ ]

3.0 DESIGN INPUT AND ASSUMPTIONS

3.1 Design Input

3.1.1 Base Code Revision and Plant Model

The base code revision is the latest MAAP4 archived revision, MAAP 4.0.5 [EPRI, 2003b]. In addition, a revision to the archived subroutine WFLOW was included in this analysis to address a finding made during the analysis, as discussed in detail in Section 2.

The base plant model is the latest PVNGS-specific plant model, or parameter file, [PVNGS, 2001] for MAAP4.

3.1.2 Analysis-Specific Plant Model Parametric Input Data

~~Table 3-1 summarizes the analysis-specific plant model parametric input data that is most~~ influential to the analysis. Some values are taken directly from the PVNGS base plant model. Others are analysis-specific changes. (Parameter input of secondary importance is not discussed here, and their values are taken from the base plant model without alternation.) [

]

[

]

[redacted]

3.1.3 Analysis-Specific Assumptions of Plant and Operator Response

In addition to plant model parametric input data, there are analysis-specific modeling assumptions of plant and operator response, which area summarized in Table 3-2. As with the parametric input data, assumptions are primarily best-estimate, but some key assumptions, which have a large bearing on RCS and containment response, are biased in a conservative manner. These are discussed here.

[redacted]

[redacted]

3.1.3.1 RCS Void Fraction for Phase Disengagement

The MAAP RCS model tracks a global primary system average void fraction. When the void fraction exceeds the value of a user input model parameter VFSEP, the gas- and liquid-phases will disengage (or separate). The phases can re-engage if the void fraction is reduced below user input model parameter VFCIRC. Phase disengagement is an important consideration because it has a substantial influence on the rate at which the RCS can depressurize.

Specifically, while the phases are engaged and under natural circulation through the coolant loops, gas and liquid are essentially in thermodynamic equilibrium. The net effect of this condition is that the break discharges at a higher mass and energy rate, which leads to a larger depressurization rate. While the phases are disengaged, gas and liquid are in thermodynamic non-equilibrium. If the phases are disengaged (but all other conditions remain the same), the break discharges at a lower mass and energy rate, which leads to a smaller depressurization rate.

The FLECHT-SEASET was a scaled integral experiment, which studied two-phase natural circulation through the RCS, including phase disengagement. For RCS configurations with inverted U-tube steam generators, phase disengagement occurred at a best-estimate void value of roughly 50%. However, there is significant uncertainty in this quantity. Sensitivity studies of MAAP with the PVNGS plant model showed that a value of VFSEP = 0.10 would disengage the phase early relative to the noted best-estimate value, leading to the noted slower depressurization rate, which is conservative for this analysis. This is demonstrated for the 3-inch LOCA in Figure 3-1. (Values below 0.10 did not result in significantly early disengagement.) Therefore, this value is used as a conservative bound, and it is paired with a corresponding value of VFCIRC = 0.05 for possible re-engagement, although re-engagement does not occur during this analysis.

[ ]

3.1.3.2 Post-LOCA Cooldown Methodology

[ ]

[ ]

3.1.3.3 Post-RAS HPSI Status

[ ]

3.1.3.4 Post-RAS CSS Status

[ ]

[ ]

3.1.3.5 Post-RAS LPSI Status

As discussed in the background in Section 1, it is virtually impossible for LPSI to experience post-RAS degradation since post-RAS restart of LPSI is not automatic and must be done by remote operator action, which carries a substantial delay relative to the automatic switchover performed by HPSI and CSS.

Therefore, it is assumed that LPSI is available in post-RAS for RCS injection and, if necessary, containment spray and long-term containment heat removal through the containment spray heat exchangers. Even though both LPSI trains are available during post-RAS operation, it is conservatively assumed for this analysis that only 1 train is aligned for post-RAS injection, and no LPSI trains are used to assist contain spray and heat removal.

4.0 MAAP CASES

This section of the MAAP analysis report (and the corresponding section of the CENTS analysis report) is organized in terms of several case series, with each series devoted to a particular combination of major boundary conditions (break location, ECCS trains, HPSI availability, etc.). (The full scope of boundary conditions is provided in Section 3.) Specific results associated with a series are discussed as part of its presentation below.

An overall summation of the analysis highlights will be conducted in Section 5.

4.1 Series 1

This series is defined by the following boundary conditions:

- Break location: Cold leg discharge
- Break size: Break diameters of ½, 1, 2, 3, 4, 5, 6, 7, 8, 9, and 10 inches
- At SIAS: 2 HPSI; 2 CSS; and 2 LPSI trains available
- At RAS: No HPSI; 2 CSS trains degraded to 25% of non-degraded flow; 1 LPSI to RCS; and 1 LPSI in reserve.

[ ]

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[ ]

4.1.1 Detailed Profile of the 3-Inch Case

A detailed profile is being provided for the 3-inch case in Series 2 since its break location is lower and therefore potentially more challenging than Series 1. A dedicated profile for the 3-inch case in Series 1 is not necessary since the same generic insights can be obtained from the profile in Series 2.

4.2 Series 2

[ ]

[ ]

4.2.1 Detailed Profile of the 3-Inch Case

[ ]

[ ]

[redacted]

[ ]

[ ]

[redacted]

[ ]

[ ]

4.3 Series 3

This series is defined by the following boundary conditions:

[ ]

Core coverage and long-term core cooling are never vulnerable, which is expected since the corresponding HPSI failure cases showed no core uncovery.

4.4 Series 4

This series is defined by the following boundary conditions:

[ ]

[ ]

Core coverage and long-term core cooling are never vulnerable, which is expected since the corresponding HPSI failure cases showed no core uncovery.

4.5 Series 5

[redacted]

~~PROPRIETARY~~

REDACTED VERSION

[redacted]

[ ]

[redacted]

[ ]

[ ]

5.0 MAAP ANALYSIS SUMMARY AND CONCLUSIONS

5.1 RCS Thermal-Hydraulic Performance

Key figures-of-merit are summarized for Series 1 cases in Table 5-1 and Series 2 cases in Table 5-2. The fundamental conclusion illustrated in these tables and discussed in detail in Section 4 is that core coverage is maintained without the use of HPSI for an extensive period between the time of RAS and the time of significant post-RAS LPSI flow, which provides long-term cooling. This is true for even the most challenging break sizes and conservative assumptions for key boundary conditions, particularly early RCS steam-water phase disengagement and a post-LOCA cooldown rate that is substantially less than the maximum allowable by emergency operating procedures.

[[redacted]]

[ ]

[ ]

5.2 Containment Thermal-Hydraulic Performance

The MAAP containment analysis in Section 4 demonstrated that the 3-inch case is generally the most challenging break size since [ ]

[ ]

[ ]

As shown in Section 4, this results in a post-RAS pressure peak in containment that is largest for the 3-inch case. However, this peak is well within the containment design basis strength.

Thus, it can be concluded that, even for the overly conservative assumption of substantial CSS degradation, post-RAS long-term containment heat removal can be achieved.

6.0 NOMENCLATURE

ADV	Atmospheric Dump Valves
BAF	Bottom of Active Fuel
CENTS	Combustion Engineering Nuclear Transient Simulation Code
CSAS	Containment Spray Actuation Signal
CSS (or CS)	Containment Spray System
ECCS	Emergency Core Cooling System
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
FAI	Fauske & Associates, LLC
HLI	Hot Leg Injection
HPSI	High-Pressure Safety Injection
JPO	Justification for Past Operations
LOCA	Loss of Coolant Accident
LPSI	Low-Pressure Safety Injection
MAAP	Modular Accident Analysis Program
MUG	MAAP User's Group
PVNGS	Palo Verde Nuclear Generating Station
RAS	Recirculation Actuation Signal
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RWT	Refueling Water Tank
SDBCS	Steam Dump and Bypass Control System
SIAS	Safety Injection Actuation Signal

SIT	Safety Injection Tank
TAF	Top of Active Fuel
TMI-2	Three Mile Island Unit 2

7.0 REFERENCES

EPRI, 2003a, "MAAP4: Module Accident Analysis Program for LWR Power Plants," Computer Code User's Manual, May 1994 – January 2003 (Updated to MAAP 4.0.5 specification in January 2003).

EPRI, 2003b, "MAAP4: Module Accident Analysis Program for LWR Power Plants – Transmittal Document for MAAP 4.0.5," FAI/03-13, January.

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"Boundary
Conditions for Palo V

PVNGS, 1992, "Palo Verde Probabilistic Risk Assessment, Section 11 Containment Analysis," Revision 0, dated April 7.

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"PVNGS Licensee
Event Report 2004-0

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"FW NSAL Distribution Request

ATTACHMENT 2-E

DAR-OA-05-3, Revision 0
Report of SBLOCA Analyses with Degraded ECCS Flow After RAS Performed for Arizona Public Service in Support of Palo Verde Nuclear Generating Station Units 1, 2, and 3
~~(Proprietary)~~
REDACTED VERSION

WESTINGHOUSE ELECTRIC COMPANY LLC

DAR-OA-05-3
Revision 0

REPORT OF SBLOCA ANALYSES WITH DEGRADED ECCS FLOW AFTER RAS PERFORMED FOR ARIZONA PUBLIC SERVICE IN SUPPORT OF PALO VERDE NUCLEAR GENERATING STATIONS UNITS 1, 2, & 3

February 2005

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Official record electronically approved in EDMS 2000.

Table of Contents

1.0 Background / Purpose

This report was prepared by Westinghouse Electric Co. for Arizona Public Service (APS) in support of Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2 & 3 . This analysis is part of a project to determine the past operability of the PVNGS units with air in the Emergency Core Cooling System (ECCS) suction lines to the containment sump.

If a Loss of Coolant Accident (LOCA) were to occur with air in the ECCS pump suction line to the sump, it is postulated that the High Pressure Safety Injection (HPSI) pump operability could be compromised due to air binding in the pump volute. This is postulated to occur at the time of the Recirculation Actuation Signal (RAS), when the HPSI and containment spray pump(s) suction shifts from the Refueling Water Tank (RWT) to the containment sump.

Two different scenarios of HPSI pump degradation have been analyzed. In the first scenario, LOCA's of various break sizes are analyzed with complete failure of the HPSI pumps after RAS initiation. Since the Low Pressure Safety Injection (LPSI) pumps de-energize at RAS, the plant operator is assumed to restart one LPSI pump to maintain Reactor Coolant System (RCS) makeup flow, in accordance with plant emergency operating procedures. In the second scenario, the same LOCA transients are analyzed with degraded HPSI pump flow, for a duration of four minutes, after which the air in the pumps has been discharged into the system and pump performance is considered to return to normal. For this second scenario, there is no operator action to restart a LPSI pump. The degraded HPSI flow condition is based upon pump performance tests performed for this project at Wylie Corporation which is documented in an APS letter to the NRC, # 102-05195-GRO/DGM/RAS, dated 12/27/2004.

Since this analysis is intended to look at past operation, best estimate conditions are assumed. This analysis is in no way considered to be part of the PVNGS licensing basis nor has it been performed to satisfy any requirements of 10CFR50.46.

The purpose of this report is to describe any detrimental effects (core uncovery) that occur or are exacerbated by the HPSI pump degradation (total loss of operability and / or degraded operation) during various small and medium break size LOCA events.

Break sizes of 1 to 10 inches in diameter are analyzed in both the cold discharge leg (CDL) and the Reactor Coolant Pump (RCP) Suction Leg (SL). Breaks smaller than one inch are not analyzed because they do not cause a Containment Spray Actuation Signal. Thus, sprays pumps are not needed and the time to RAS is sufficiently long to allow a plant cooldown and shift to shutdown cooling. For these small breaks, pressurizer level is regained without RCS water levels dropping below the level of the hot legs. Breaks greater than 10 inches in diameter are not analyzed because RCS pressure is well below the LPSI pump shutoff head at the time of RAS. Therefore, flow from the LPSI pump, restarted by the operator after RAS, is greater than normal HPSI pump flow from two pumps. Thus, break sizes greater than 10 inches in diameter are not considered limiting. Only the two cold leg break locations are analyzed because any breaks in the hot leg would allow venting of steam produced in the reactor core directly to the containment, without need for loop seal clearing or draining. RCS depressurization occurs without depressing water level below the top of the core. Thus, cold side breaks are limiting regarding core uncovery. A sensitivity case with a break in the pressurizer was performed to verify the limiting nature of cold leg breaks.

This report was prepared according to Westinghouse Procedure WP 4.25, Rev. 2, 11/30/04, and is supported by Westinghouse Calculation Note CN-OA-05-1, Rev. 0, dated 02/11/05.

2.0 Code Description

The Westinghouse CENTS computer code has been utilized for this analysis.

[redacted]

[ ]

3.0 Case Descriptions & Input Parameters

3.1 COMMON INITIAL CONDITIONS AND PLANT PARAMETERS

3.1.1 Initial Plant Conditions

[ ]

3.1.2 ECCS Parameters

The initial ECCS conditions and assumptions are the same for all cases except the sensitivity cases. These parameters are as follows.

[ ]

3.2 BREAK PARAMETERS

[ ]

3.3 CORE DECAY HEAT

[ ]

[ ]

3.4 CONTAINMENT SPRAY PUMPS

[ ]

[ ]

3.5 OPERATOR ACTIONS

Operator actions are in accordance with the APS emergency operating procedures. In particular, the Loss of Coolant Accident Procedure [ ], was used to determine the simulated operator responses to the transient. The actions taken are similar for all the cases analyzed, though the timing of some actions is different for each case. The actions are summarized as follows:

[ ]

3.6 SENSITIVITY CASES

Seven sensitivity cases have been analyzed to support this analysis. The sensitivity cases are intended to show the effects on overall case acceptability for those parameters which play an important role in the transient and could vary in some significant way from the values chosen for the various series of cases. Details are discussed below.

[redacted]

[redacted]

4.0

[ ]

5.0 Case Results

[ ]

[redacted]

[ ]

[ ]

5.2 DISCUSSION OF INDIVIDUAL CASE RESULTS

In the discussion below, the failed HPSI and degraded HPSI cases are discussed together. Prior to RAS these cases are identical. After RAS, it is useful to compare how the relative ECCS flows affect the remainder of the events.

A review of the figures showing ECCS flow provides some perspective on the overall effect of degraded HPSI flow for four minutes, after RAS. As an example, for the [ ] CDL break with degraded HPSI, Figure 7.2.3.3 shows the ECCS flow. RAS occurs shortly after [ ] seconds. A visual review of the degraded HPSI flow indicates that the depleted flow is a very small portion of the integrated flow over the course of the event. It would be expected to have very little effect on event results. This fact is supported by the Sensitivity case [ ] which show that nominal vs. degraded HPSI flow does not significantly change case results.

[ ]

5.2.1 Series 1 & 2 Cases: Cold Discharge Leg (CDL) Breaks

CDL -1

[ ]

[redacted]

[ ]

5.2.2 Series 3 & 4 Cases: Suction Leg (SL) Breaks

[ ]

[ ]

5.2.3 Series 5 Cases: Sensitivity Cases

[redacted]

[ ]

5.3 CASE SUMMARY

[ ]

6.0 Conclusions

The series of cases described above show that degraded HPSI flow caused by the air in the ECCS sump suction line will not lead to situations where core uncovery would occur. Two cases with total HPSI pump failure at RAS led to some partial core uncovery for an extended period of time, due to a depletion of RCS inventory. Those were the 3" and 4" CDL breaks. **There were no cases with degraded HPSI pump flow which had any partial core uncovery associated with the degraded ECCS flow.**

There were some additional cases, both failed and degraded HPSI flow cases, that showed short periods of partial uncovery due to loop seals filling and clearing; however, this phenomenon is expected for both CDL and SL breaks and is not due to the degraded flow in the ECCS system. This was verified by Sensitivity Case 7.

7.0 Figures

7.1 SERIES 1: CDL BREAKS, FAILED HPSI AFTER RAS

7.1.1 CDL-1

[ ]

[redacted]

[ ]

7.1.2 CDL-2

[ ]

7.1.3 CDL-3

[ ]

[redacted]

[redacted]

7.1.4 CDL-4

[ ]

7.1.5 CDL-5

[redacted]

[redacted]

[ ]

7.1.6 CDL-6

[ ]

[redacted]

7.1.7 CDL-7

[ ]

[ ]

[ ]

7.1.8 CDL-8

[ ]

[redacted]

7.1.9 CDL-9

[redacted]

[redacted]

7.1.10 CDL-10

[redacted]

7.2 SERIES 2: CDL BREAKS, DEGRADED HPSI AFTER RAS

7.2.1 CDL-1 DH (Case not Required)

7.2.2 CDL-2 DH (Case not Required)

7.2.3 CDL-3 DH

[redacted]

7.2.4 CDL-4 DH

[redacted]

7.2.5 CDL-5 DH

[ ]

[[redacted]]

7.2.6 CDL-6 DH

[ ]

[redacted]

7.2.7 CDL-7 DH

[ ]

[ ]

[redacted]

7.2.8 CDL-8 DH

7.2.9 CDL-9 DH

7.2.10 CDL-10 DH

7.3 SERIES 3: SL BREAKS FAILED HPSI AFTER RAS

7.3.1 SL-1

[ ]

[redacted]

7.3.2 SL-2

[redacted]

[ ]

7.3.3 SL-3

[redacted]

[ ]

[redacted]

7.3.4 SL-4

[ ]

[ ]

[ ]

7.3.5 SL-5

7.3.6 SL-6

[redacted]

7.3.7 SL-7

[redacted]

7.3.8 SL-8

[redacted]

[ ]

7.3.9 SL-9

[ ]

7.3.10 SL-10

[ ]

7.4 SERIES 4: SL BREAKS, DEGRADED HPSI AFTER RAS

7.4.1 SL-1 DH (Case not Required)

7.4.2 SL-2 DH (Case not Required)

7.4.3 SL-3 DH

[ ]

[redacted]

[redacted]

7.4.4 SL-4 DH

[ ]

7.4.5 SL-5 DH

7.4.6 SL-6 DH

[redacted]

7.4.7 SL-7 DH

[ ]

[[redacted]]

7.4.8 SL-8 DH

7.4.9 SL-9 DH

[ ]

[redacted]

7.4.10 SL-10 DH

[redacted]

7.5 SERIES 5: SENSITIVITY CASES, FAILED HPSI AFTER RAS, 2" SL BREAK

7.5.1 SL-2 75F CD

[ ]

[redacted]

7.5.2 SL-2 1 HPSI

[ ]

[redacted]

7.5.3 SL-2 SIT Gamma = 1

[ ]

[ ]

7.5.4

7.5.5

[redacted]

[ ]

[redacted]

7.5.6 CDL-6

[redacted]

7.5.7 CDL-8

[redacted]

ATTACHMENT 2-F

**13-NS-C074, Revision 0,
Significance Determination of Containment Sump
Air Entrainment**

DOCUMENT NUMBER					
13-NS-C074					
Q		QAG		NQR	X

PALO VERDE
NUCLEAR GENERATING STATION

Title / Description: **Significance Determination of Containment Sump Air Entrainment**

REV. NO.	REVISION DESCRIPTION	Preparer / Date	RE / Date	Second Party Verification / Date	Mech. / Date	Civil / Date	Elec. / Date	I & C / Date	Independent Verification / Date	Other (Specify Org.) / Date
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△ 0	Document Electronically Available Yes [x] No [ ]	Digitally signed by: Lindquist Jr, Robert C (Z60942 Date: 02/15/2005 07:52:56 Reason: I am the author of this document Location: PVNGS	Digitally signed by: Lindquist Jr, Robert C (Z60942 Date: 02/15/2005 07:53:18 Reason: I am the author of this document Location: PVNGS	N/A	N/A	N/A	N/A	N/A	Digitally signed by: Sowers, Gerald W (Z90174) Date: 02/15/2005 07:58:55 Reason: approved per telecon for L. Bullington Location: PVNGS	N/A

CROSS DISCIPLINE REVIEW

1.0 Introduction

The purpose of this study is to document Palo Verde's Phase III significance determination of the containment sump air entrainment condition. Specifically, this condition is the lack of water upstream of the sump check valves to the inside containment sump isolation valves. When the sump isolation valves are opened by the Recirculation Actuation Signal (RAS), that air does not have a chance to escape back to the containment atmosphere, but is swept along with the sump water to the suction piping of the ECCS and Containment Spray pumps.

The NRC's significance determination showed this condition to be a YELLOW finding (delta-CDF between 1E-5 and 1E-4/yr). This study will show that it is a WHITE finding (delta-CDF between 1E-6 and 1E-5/yr).

Section 2 will first compare and contrast the assumptions the NRC used in their Phase 3 analysis using their SPAR model vs. modeling assumptions in the PVNGS PRA. Section 3 presents the methodology employed by PVNGS for our Phase 3 analysis. Section 4 shows the results of our analysis. The appendices present background material to support our analysis.

The PRA model used for the analysis is as documented in Engineering Study 13-NS-C029 Rev 13, Ref. 1, with changes as noted in Section 2.

2.0 Assumptions and Initial Conditions

2.1 Comparison of NRC and PVNGS Assumptions

Prior to presenting the PVNGS analysis, it is useful to see the differences between the NRC SPAR model analysis and the PVNGS PRA.

NRC	PVNGS
HPSI and Cont Spray fail on RAS	HPSI only fails for breaks 2" or less; CS not affected
Operators recover HPSI by venting	No HPSI recovery; venting not credited
Operators recover CS by venting	N/A since CS not affected
No alternative success path for high pressure sump recirculation or containment spray	Cool-down and depressurization for SIT injection and low pressure sump recirculation
Consequential RCP seal LOCA for transients and Loss of Off-Site Power	RCP seal LOCA no longer modeled due to low leak rates and insignificant contribution even for catastrophic pump failures
Consequential PSV lifting not included in LOOP/SBO	Consequential PSV lift is modeled for SBO

Recirculation Actuation Signal occurs, and it is discovered that high pressure sump recirculation is not functioning. Since this constitutes loss of a safety function, the operators are directed to the Functional Recovery Procedure, Ref. 8. The HRA 1RC-SBLOCA-L-2HR was examined to ensure it reflects the proper diagnosis and implementation timing. This HRA had been based on avoiding containment failure. Basing it on avoiding core damage did not change its value.

Several simulator runs were done as part of this investigation. They provided confidence that the operators would properly diagnose the LOCA, commence cool-down expeditiously, and then properly respond to the loss of high pressure sump recirculation. The value of 1RC-SBLOCA-L-2HR is 6.70E-3 with an error factor of 5. Thus the median value, which many references, including NUREG-1278 suggest using, is a factor of 1.7 lower. Therefore the value used in the PVNGS PRA model is believed to be conservative and robust.

It should also be pointed out that, whereas the HRA is part of the accident sequence and on the HPSI success path, the NRC's SDP evaluation applied a non-recovery factor (value of 0.24) to the results of their analysis to credit venting of piping as a HPSI pump recovery strategy. This action would be done under much greater stress and involve ex-control room actions. The two actions are thus very different and should not be compared.

2.5 Discussion of RCP Seal Leak/LOCA

The NRC's SPAR model includes RCP seal leak/LOCA. This was removed from the PVNGS PRA model because of CE/Westinghouse development of a new failure model, Ref. 9, which is nearing NRC approval, and due to the fact that the CE-KSB pumps used at Palo Verde have a very tight clearance between the pump seal package and shaft, such that only 17gpm leakage would result if all three seal stages on a pump failed (Ref. 4). All four pumps together would result in a leak rate within the capacity of two charging pumps. Sensitivity studies done as part of the impact that removed the seal modeling (2001-216) showed that using the CE/W model with conservative assumptions resulted in no significant increase in risk from failed RCP seals.

2.6 Internal Fires

The mitigating event trees for internal fires use internal event transient trees as their basis. The effect of a potential partially-stuck-open Pressurizer Safety Valve is also quantified as part of this analysis.

3.0 Solution Methodology

3.1 Determination of New Baseline Recovered CDF and LERF Values

The PRA model was re-quantified for internal events following introduction of the changes intended to be permanent. Those changes are confined to the Small LOCA analysis as noted in Section 2.2. The addition of HPSR failure mitigation for transient events used for the PSV LOCA sensitivity analysis was not included. The new baseline values are:

$$CDF_{base} = 1.34E\text{-}5/yr$$
$$LERF_{base} = 1.57E\text{-}6/yr$$

4.0 Results and Conclusions

4.1 Small LOCA

The change in CDF and LERF were determined given that LOCAs of two inches or less equivalent diameter cannot be mitigated using HPSR. CDF_{base} and $LERF_{base}$ are the values reported in Section 3.1. $CDF_{no\text{-}hpsr}$ and $LERF_{no\text{-}hpsr}$ are from Appendix B:

$$\text{Delta CDF} = CDF_{no\text{-}hpsr} - CDF_{base}$$
$$= 1.79\text{E-}5/yr - 1.34\text{E-}5/yr$$
$$= 4.5\text{E-}6/yr$$

$$\text{Delta LERF}_{no\text{-}hpsr} - LERF_{base}$$
$$= 1.57\text{E-}6/yr - 1.57\text{E-}6/yr$$
$$= 0.0/yr.$$

These results are sufficient to show that LERF is not affected by the inability of HPSR to address small LOCA. Thus it will not be considered any further in this study. This is consistent with the NRC's analysis.

4.2 PSV Partial Open Failure

Appendix B shows the determination of risk increase for PSV failing open assuming HPSR is not capable of its mitigation. The change in risk is:

Delta CDF for Internal Events = 2.4E-7/yr
Delta CDF for Internal Fires = 1.8E-6/yr

Fire is dominant because there are so many fire event trees which contain the PSV failing open, and because many have boundary conditions that disable some mitigating equipment.

4.3 External Events

4.3.1 External Flooding, Transportation and Nearby Facility Accidents, Sandstorms and Extreme Heat

External flooding, transportation and nearby facility accidents, sandstorms and extreme heat fall in the category where plant design is adequate to prevent a plant trip or the frequency of a plant trip was negligible when compared with other plant trip sources. Additionally, none of these events would have an impact on the availability or the reliability of mitigation equipment used in the same accident sequences as high pressure recirculation, nor would they create a new accident sequence that would result in the need for high pressure recirculation.

Therefore, there is no or a negligible increase in risk due to external flooding, transportation and nearby facility accidents, sandstorms and extreme heat given the "dry containment sump" deficiency.

4.3.4 Impact of External Events - Conclusion

Upon review of the external events potentially impacting PVNGS, most external events were either found to be negligible in comparison to the corresponding internal events or occurred at a frequency judged to be high enough to be included as part of the internal initiating event frequency. External events are not expected to impact the reliability or availability of mitigating equipment used to mitigate the initiators from the internal events evaluation. External events are not expected to create a new sequence that would result in the need for recirculation.

The contribution to risk from the "dry containment sump" condition due to external events is dominated by seismic events and is estimated to be 4.72E-07/year.

4.4 Internal Flooding

Internal flooding does have the potential to impact initiators loss of condenser vacuum (IECONDVAC) and loss of nuclear cooling water (IENCW), where the result of a plant trip could lead to a stuck-open primary safety valve (transient induced LOCA requiring high pressure recirculation). Upon review of the PVNGS internal events event trees, IECONDVAC and IENCW are two transients that are subject to internal flooding and whose event tree includes the potential for a stuck-open primary safety valve.

As addressed in Section 2, loss of RCP seal cooling leading to RCP seal failure would not result in a small break LOCA. The RCS leakage from the failed RCP(s) seals would be within the capacity of the charging system. Hence, loss of plant cooling water or loss of nuclear cooling water events would not impact the subject performance deficiency due to a failed RCP seal.

Turbine building flooding is not included in the PVNGS PRA, since its contribution is negligible compared with other events that could cause a plant trip. To be consistent with the NRC Phase 3 review, the same bounding frequency (9.6E-04/year) for internal flooding will be used with IECONDVAC and IENCW. The change in core damage frequency (delta-CDF) would then be the difference between the internal events model that includes the partially stuck open primary safety valve and the internal events baseline model. The impact of a partially stuck open primary safety valve and impact upon HPSI recirculation is discussed in Appendix B. Both models will include an additional 9.6E-04/year (due to flooding) for both IECONDVAC and IENCW.

Model Configuration and Calculation (including modified IECONDVAC and IENCW for flooding)	**IE value with flood**
IECONDVAC	4.50E-02/year
IENCW	9.88E-03/year
	CDF (per yr)
PSV without HPSR with IE Flood	1.290E-5/yr
PSV without HPSR	1.289E-5/yr
(delta-CDF)	**1E-8/yr**

References

1. Engineering Study 13-NS-C029 Rev.13, *Interim PRA Change Documentation*
2. NRC Special Inspection Report, Letter EA-04-221 from Arthur T. Howell III to Greg R Overbeck, dated January 5, 2005
3. Significant CRDR 2726509, *Safety Significance Evaluation of ECCS Containment Sump Voided Piping*
4. *Seal Flow and Leakage Retrofit Report for Reactor Coolant Pumps at the Palo Verde Nuclear Generating Station (SBP1 Seal Type RCR950-B3)*, Sulzer Bingham Pumps, July 1996
5. PRA Model Impact 2005-2
6. Westinghouse Report DAR-OA-05-3 Rev 0, *Report of SBLOCA Analyses with Degraded ECCS Flow After RAS Performed for Arizona Public Service Company in Support of Palo Verde Nuclear Generating Stations Units 1, 2 & 3*, January 2005
7. Emergency Operating Procedure 40EP-9EO03 Rev 17, *Loss of Coolant Accident*
8. Emergency Operating Procedure 40EP-9EO09 Rev 22, *Functional Recovery*
9. WCAP-16175-P, *Model for Failure of RCP Seals Given Loss of Seal Cooling in CE NSSS Plants*, January 2004

Actuation setpoint. HPSI is successful if at least one pump operates to inject water from the RWT through at least three of the four available injection pathways.

2.3 Secondary Heat Removal (SGHR)

Heat removal through the steam generators is required to achieve core cooling for all Small LOCAs. This is the basis for the division between small and medium LOCAs. With successful HPSI, secondary cooling success is at least one AFW pump supplying either of the two steam generators and at least one ADV or steam bypass valve steaming. (A different success criterion for secondary cooling is used in the DPRS1 function discussed below.)

2.4 Operators Cool Down and Depressurize the Plant (DPRS3)

This function consists of a single basic event, which is a HRA. However, two different HRAs are used depending on the sequence. Where HPSI is successful, the operators are following the LOCA emergency procedure (Ref. 7) and have a considerable amount of time to diagnose and execute plant cool-down, although it is expected to commence expeditiously; i.e., within about 30 minutes. However, if HPSI fails, the operators are directed to the functional recovery procedure (Ref. 8). This is a much more urgent situation, because considerably less time is available due to the high rate of blowdown with no makeup. Success in either case is the proper diagnosis, commencement and execution of cooling down and depressurizing the RCS.

2.5 High Pressure Safety Recirculation (HPSR)

After depletion of the RWT, a Recirculation Actuation Signal (RAS) is generated, which opens the containment sump isolation valves to supply suction to the Containment Spray pumps and the HPSI pumps. (LPSI pumps are shut down by the RAS, but may be restarted if needed.) HPSR success is opening of the sump isolation valves and closure of either the RWT outlet check valves or outlet isolation valves, such that sump water is not diverted back to the RWT.

2.6 Shutdown Cooling (SDC1)

If the accident proceeds as expected, the operators will align and start shutdown cooling in order to bring the reactor to a cold shutdown and depressurized condition, which essentially terminates the LOCA, thus minimizing the need for makeup. One SDC loop, which consists of a LPSI pump taking suction from a hot leg and pumping it through a Shutdown Cooling Heat Exchanger and back to the RCS through one of two available cold leg injection points is required for success.

2.7 Depressurize for Low Pressure Safety Injection (DPRS1)

Low Pressure Safety Injection, LPSI, can be used to replace a failed HPSI or HPSR function. This requires that the operators cool down the plant further than would be necessary for the use of Shutdown Cooling. The operator actions themselves are contained in function event DPRS3. DPRS1 consists of the plant equipment required to achieve this, which are one AFW pump supplying water to both steam generators; one ADV on each steam generator or steam bypass valves; also all four Safety Injection Tanks (SITs) are required for inventory control; and finally for the HPSI failure sequence, one LPSI pump supplying injection to at least one cold leg.

secondary cooling (both SGs) and SITs for makeup are successful, along with a successful restart of a LPSI pump with successful injection; however, LPSI fails in the recirculation mode.

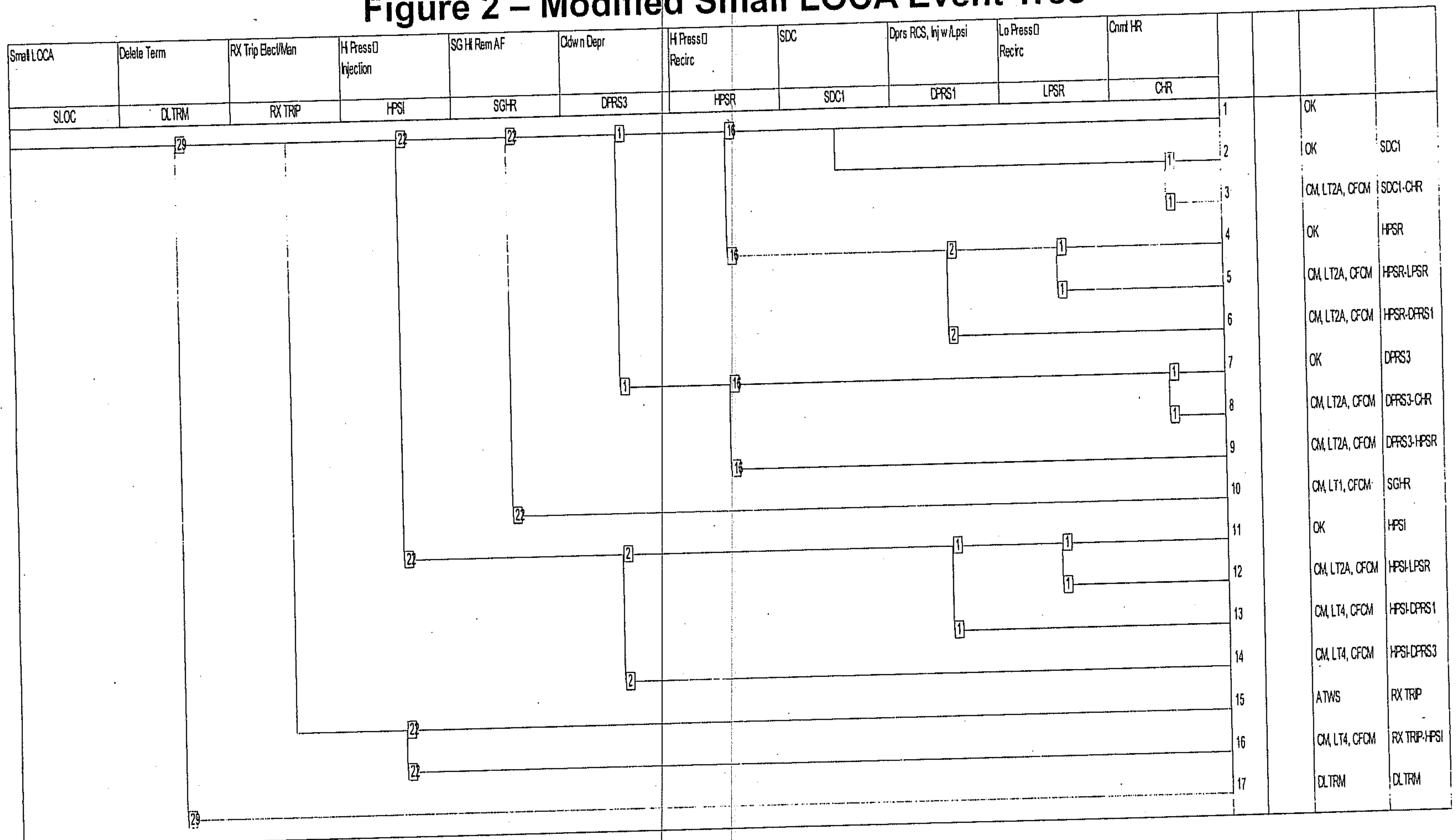
3.8 **Sequence 13:** Successful reactor trip; HPSI fails; operators cool down and depressurize the reactor per the functional recovery procedure; secondary cooling (both SGs), SITs or LPSI for makeup, fails.

3.9 **Sequence 14:** Successful reactor trip; HPSI fails; operators fail to diagnose or execute plant cool-down.

3.10 **Sequence 15:** Reactor trip fails; HPSI is successful. This leads to an ATWS sequence that can be mitigated in a separate event tree.

3.11 **Sequence 16:** Reactor trip fails; HPSI fails. This leads to an ATWS sequence that is assumed cannot be mitigated, so leads directly to core damage.

Figure 2 – Modified Small LOCA Event Tree



Appendix B –PSV Fail-Open Mitigation

1.0 Model Changes

To allow an alternate success path for HPSR failure following a PSV failing to close, top logic must be altered for the HPSR function event, such that it includes the systems and operator actions necessary to effect a rapid cool-down and depressurization to allow SITs to inject and ultimately sump recirculation via a LPSI pump, which would have to be restarted (RAS shuts down the LPSI pumps).

The current model was used as a starting point and so does not contain the changes made to evaluate Small LOCA. However, Small LOCA modeling may be used as a guide, since this alternate success is included for the HPSI failure sequence. The top logic tree GTLINJECT, Figure 2, includes the systems and HRAs needed. The effect of a PSV sticking open is dominated by water relief sequences where AF has failed and Alternate Feedwater has succeeded. Thus calling in AF in the HPSR function event would always lead to failure. However, since Alt Feedwater succeeded, not only is secondary cooling successful, but the operators must also have been successful in cooling down and depressurizing the plant. With SG pressure low enough to allow use of condensate pumps to feed, primary pressure allowing 20F subcooling would be about 500 psia. This is low enough to have SITs injecting. Thus the logic under gate GTLINJECT-1 is not required. Only that under GTLINJECT-4 is required. Also, LPSI injection is not required, but low pressure sump recirculation is. Therefore, gate GLR-4-4 is substituted for GLI-4-4.

However, two minor non-conservatisms result by not including the AFW input and the depressurization HRA:

1) Both would still be needed for the PSV steam relief sequences and for the SBO water relief sequence; these sequences were quantified and the error was found to contribute much less than one percent of CDF. The PSV steam relief sequences in the fire model were also checked and all were much less than one percent.

2) The success criterion for Alternate Feedwater in the SGHR_CD function event is one pump to <u>either</u> steam generator, whereas the success criterion in the HPSR function event should be one pump to <u>both</u> steam generators. The ALTFW top gate was solved both as an AND gate and as an OR gate. The difference was about 0.4 percent.

Model changes are as follows:

- Create new gate and fault tree HPSR-FAILS-1 as shown in Figure 1.
- Add Function Event Alternative 25 to Function Event HPSR. Alternative 25 uses gate HPSR-FAILS-1. There is no boundary condition set applied.
- This alternative provides for mitigation using SITs, and LPSR and is assigned to HPSR events in trees that include PSV, both in internal events and fire mitigation event trees.
- Gate HPSR-FAILS-1 has two variations. The first uses the nominal fault tree input for the HPSR function (GHR-7-8). The second has the gate GHR-7-8 set to TRUE.

Significance Determination of Containment Sump Air Entrainment

Figure 1

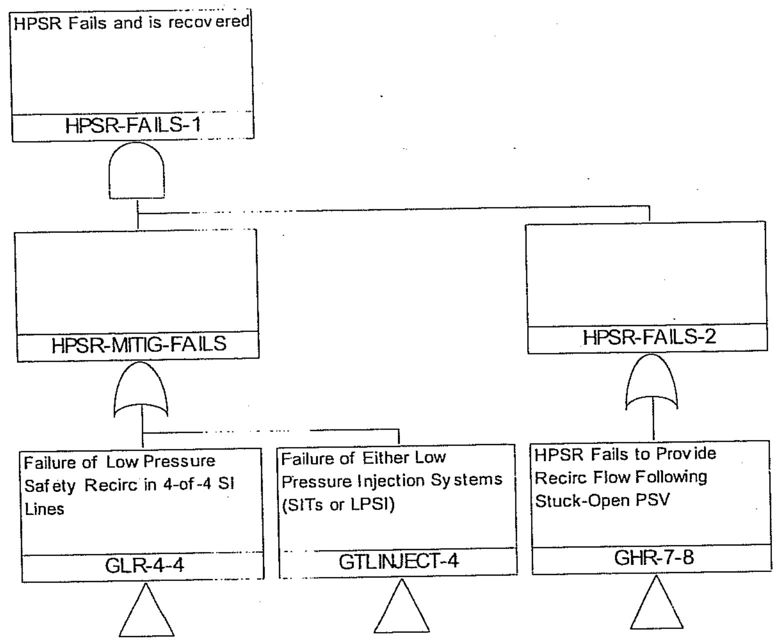
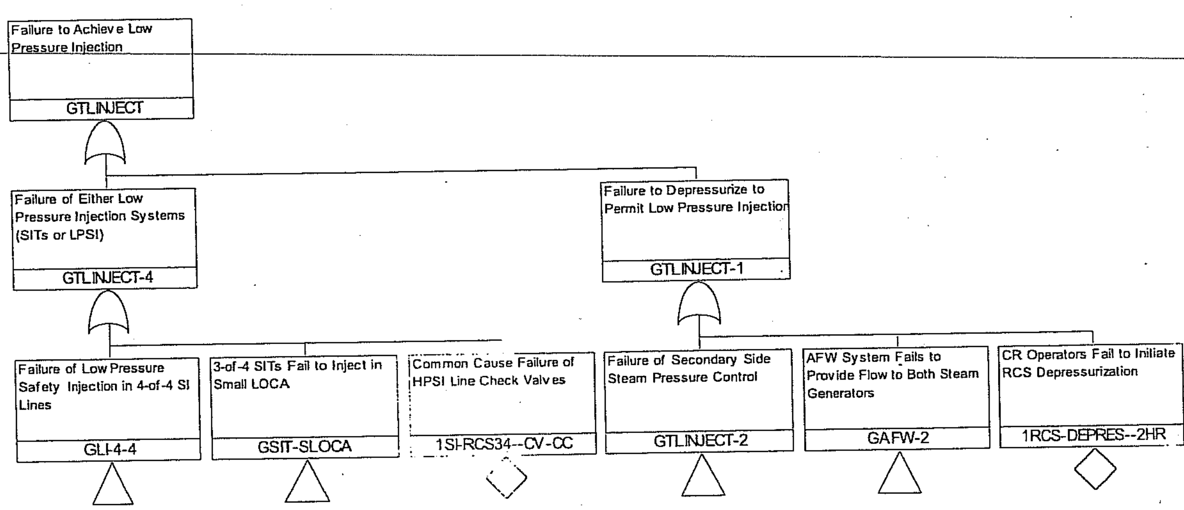


Figure 2



Appendix C - NRC Phase 3 Review for External Events, Assumptions and Conclusions

1.0 General Criteria for Evaluating External Events

1. For the subject performance deficiency to cause an increase in plant risk from an external initiator, the initiator had to do one of three things:
   a. Cause an increase in the likelihood of an internal event affected by the subject performance deficiency
   b. Affect the reliability or availability of mitigating equipment used to mitigate the initiators from the eternal event evaluation; or
   c. Cause a new sequence that would result in the need for recirculation.

2.0 Transportation Incidents, External Fires

Assumptions

1. The impact upon transients from the subject performance deficiency is the potential to induce a stuck open safety valve.
2. The impact upon loss of offsite power from the subject performance deficiency is the potential to induce a RCP seal failure.
3. Events that were initiated and remained outside of the plant, would not be expected to cause a plant system pipe break.
4. Likelihood of having an external event occur simultaneously with a major pipe break was considered to be negligible.
5. The potential for transportation incidents or external fires to induce a stuck open safety valve would be negligible.

Conclusions

1. Transportation incidents and external fires would only affect plant transients and loss of offsite power.
2. Since transportation incidents and external fires are rare events in comparison to equipment related and weather related events, the change in initiator event likelihood would be very low.
3. The increase in risk associated with the subject performance deficiency was negligible with respect to transportation events and external fires.

3.0 External Flooding

Assumptions

1. Because of the topography of the site and nature of the desert, all external floods will drain or quickly be absorbed by the environment.
2. External flooding had no expected affect on total risk

Conclusions

1. Seismic events with a magnitude greater than the review level earthquake were expected to occur at a frequency of 3.0E-05/year.
2. All Seismic 1 structures were built to withstand this review level earthquake with appropriate engineering margin.
3. Frequency of transients and loss of off-site power events would be several orders of magnitude higher that that of severe seismic events.

Assumptions

1. The normal engineering factors and resulting rigidity that were built into the Palo Verde units were sufficient to protect the plant from all but the most severe of seismic events.
2. The analyst assumed that the likelihood of a seismic event causing an initiator by affecting Seismic Category 1 equipment was low and that the change in risk associated with the subject finding would be negligible. This is based on the assumption that a seismic event large enough to cause a major pipe rupture would likely result in core damage.
3. Because of the low frequency of seismic events and the low likelihood that seismic events would cause a loss of mitigating equipment, combined with the high likelihood of a transient or loss of off-site power, the change in initiating event likelihood (added: due to seismic events) would be very low.
4. The EPRI seismic margins assessment evaluation was conservative and there was a probability that the reactor coolant system would survive earthquakes larger than the review level earthquake.
5. Seismic induced small-break loss of coolant accident could result at a rate of 3.0E-05/year.

~~7.0 Internal Fire~~

Assumptions

1. The probability of an internal fire causing a loss of nuclear cooling water was extremely low, based upon normal separation.
2. Internal fires could not cause a medium or large-break loss of coolant accident.
3. Probability of an internal fire causing a loss of offsite power was extremely low, because of equipment separation inside the plant.
4. The probability of an internal fire resulting in a stuck-open safety relief valve that was not recoverable, that the relief valve caused a plant transient, and the that the operators were unable to take the plant to cold shutdown conditions prior to recirculation was judged extremely low.
5. Internal fire events happen frequently enough that the impacts of these events are already incorporated into the initiating event frequency for a transient.
6. Internal fire could result in the complete loss of the plant cooling water system. However, the effect of this event would be no different if it were caused by an internal fire than it would be if it were initiated by equipment related problems.

Conclusions

1. The effect of internal fires was considered to be negligible with respect to the dominant transient sequences.

1. The total effect of external initiators on the change in core damage frequency from plant transients related to the subject performance deficiency was determined to be negligible.

9.3 Loss of Offsite Power

Assumptions

1. Many of the external initiators appear to cause an increase in the initiating event likelihood for a loss of offsite power.
2. High winds and certain other external events have occurred at such a high rate throughout the industry, the analyst believes they are well represented in the published loss of offsite power initiating event frequencies.
3. Internal fires were not likely to increase the probability of a loss of offsite power significantly because of the normal separation of plant equipment and because the published initiating events frequencies would include the contribution from large switchyard fires.

Conclusions

1. The frequency of seismic events, external fires, and transportation issues is low compared to equipment and human error loss of offsite power events. Since the frequency for these events is so low, the impact from these external initiators is considered negligible.
2. Since high winds and certain other external events have occurred at such a high rate throughout the industry, their effect on risk related to the subject performance deficiency is fully quantified during the internal events analysis.
3. The total effect on the change in CDF from a loss of offsite power related to the subject performance deficiency was determined to be negligible.

9.4 Loss of Plant Cooling Water System

Assumptions

1. The effect from the subject performance deficiency on a loss of plant cooling water initiating event would be an increase in the initiating event frequency from an internal flood or an internal fire affecting all system pumps.
2. The increase in risk from internal floods is assumed to be bounded by the change in CDF from the equipment related initiator (1.22E-09).
3. The probability of a large oil fire causing a loss of plant cooling water system initiating event was at lease an order of magnitude lower because the fire had to initiate, cause spilling of oil, and spread rapidly enough to damage system equipment, but not so rapidly that it would extinguish before causing a loss of the entire system.
4. The increase in CDF from an internal fire would be not greater than the internally initiated change in risk. However, because of uncertainties in the data and to ensure that the risk is appropriately bounded, the analyst assumed that the change in CDF could be as much as 10 times higher than for internally initiated events alone (1.22E-08).

Quantification

Change in CDF_{LOPCW} = flood contribution + fire contribution

Impact Review

ChangeID	Change_Description	Disposition
2000-85	PSA Peer Review Observation AS-02 states that discussion of internal flooding evaluation results should be added to the Initiating Event study.	The internal flooding analysis is currently in progress. Internal flooding is being addressed in this study.
2000-86	PSA Peer Review Observations SY-03 and SY-05 find the existing documentation is difficult for external observers to link references to individual assumptions and key inputs to the model (reliabilities, probabilities, basic events and gates).	There is no impact on Total CDF or LERF.
2000-87	PSA Peer Review Observations DA-03 and DA-06 state that the process used to group components together for data development be documented.	This is a documentation enhancement and there is no impact on Total CDF or LERF.
2000-91	PSA Peer Review Observation QU-01 found the documentation of quantification difficult to follow and recommended adding a section covering the delete term logic and recovery pattern table to 13-NS-B67.	This is a documentation enhancement and there is no impact on Total CDF or LERF.
2000-92	PSA Peer Review Observations DA-04 and DA-05 advocate use of newer 1998 INEEL data for determining common cause.	This is in progress. There is not expected to be a significant impact to CDF or LERF. There would be virtually no impact to the delta-CDFs being determined for this application.
2001-246	Add the GTG control power and diesel start batteries to the model, they are not tested by the GTG monthly start or 6 month loaded run, they are required for success in a blackout, and have a different test interval than the rest of the GTG.	There would be virtually no impact to the delta-CDFs being determined for this application.
2001-247	Establish an engineering reference document for operation of electric AF pump on one GTG and operation of HPSI and AF pump on paralleled GTGs as currently assumed for success criteria in 13-NS-B061. Update GTG failure rates given new isochronous testing.	This impact is resolved. It resulted in less than 1% change to CDF and LERF. There would be virtually no impact to the delta-CDFs being determined for this application.

ChangeID	Change_Description	Disposition
2002-87	Change LOCA frequencies to the values in SECY-04-0060 (issued April 13, 2004). The IE values from NUREG/CR-5750 (in Rev4) may be under estimates. The CRDM Nozzle events at Summer & Davis Besse caused this re-evaluation.	No significant impact is expected when the Reg Guide is finally issued. There would be virtually no impact to the delta-CDFs being determined for this application.
2003-13	ERIN fire model peer review F&O 1-1; level D	Documentation only. No impact.
2003-14	ERIN fire model peer review F&O 1-2. Level C	This has been resolved. Documentation only. No impact.
2003-15	ERIN fire model peer review F&O 2-2. Level C	Fire frequencies to be updated as part of a periodic process. No significant change is expected. There would be no impact to the delta-CDFs calculated for this application.
2003-17	ERIN fire model peer review F&O 3-3. Level D.	Documentation only. No impact.
2003-173	Clarify reference to fire brigade in memo S-13-NS-C053. Current wording implies an implicit crediting of fire detection and brigade response, which is misleading.	This is resolved. Documentation only. No impact.
2003-174	Investigate the modeling assumptions of JCDNPV0200**VALVEX with respect to common mode failure of all 3 CD Pumps.	This is resolved. Less than 1% change to CDF and LERF. There would be no impact to delta-CDFs calculated for this application.
2003-18	ERIN fire model peer review F&O 3-4. Level D.	Any error is expected to be small and in the conservative direction. There would be no impact to the delta-CDFs calculated for this application.
2003-20	ERIN fire model peer review F&O 4-2. Level C	This is resolved. There was no measurable effect on CDF or LERF.
2003-22	ERIN fire model peer review F&O 4-4. Level D	This is resolved, Documentation only. No impact.
2003-23	ERIN fire model peer review F&O 4-6. Level B	This is resolved. Documentation only. No impact.
2003-24	ERIN fire model peer review F&O 4-7. Level C	This is resolved. There was no measurable effect on CDF or LERF.
2003-27	ERIN fire model peer review F&O 5-2. Level C	This is resolved. There was no measurable effect on CDF or LERF.

ChangeID	Change_Description	Disposition
2003-55	Risk category for EQIDs/Test numbers evaluated by EOOS needs revision based upon PRA review.	EOOS issue only.
2003-60	Remodel compartment 56B using proper FIGNs and modify the general description to properly reflect the scenarios. There should be two fixed ignition sources. Currently, the FIGNs for compartment 47B are used.	This is resolved. Less than 1% change to CDF and LERF. There would be no impact to delta-CDFs calculated for this application.
2003-64	Remove logic and basic events associated with fire suppression for fire zones 5A and 5B, ESF Switchgear Rooms. Impact 2003-41 removed the suppression function event from those two event trees.	Documentation only. No impact.
2004-125	The NIRM PROC document is 40EP-9EO10 R032 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO10 R031	Documentation only. No impact.
2004-132	S/U Transformer SWYD breakers OOS (both) incorrectly fails power to NAN-S05/6 even when loads are transferred to the Alternate S/U Transformer	This only affect out-of-service modeling. No impact to this application.
2004-133	Revise the IELOOP value (in study 13-NS-C004) based on the new EPRI study TR-1009889 dated April 2004.	This is in progress. Value changes are expected to lead to an increase in CDF and LERF. However, for this application, LOOP is not a significant contributor, and delta-CDFs would not be impacted.
2004-134	EOOS System Alignment congifuration file and PRA Model do not allow proper settings for S03B/S04B FBT and NBNS01C Blocking.	EOOS issue. No impact.
2004-135	The NIRM DWG document is 01-M-CDP-0001 R016 this revision is later than the revision indicated by Risk Spectrum which is 01-M-CDP-0001 R015	Document Revision change only. No impact.
2004-136	The NIRM DWG document is 01-E-PGA-0003 R007 this revision is later than the revision indicated by Risk Spectrum which is 01-E-PGA-0003 R006	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-148	The NIRM PROC document is 41AL-1RK1B R034 this revision is later than the revision indicated by Risk Spectrum which is 41AL-1RK1B R033	Document Revision change only. No impact.
2004-149	The NIRM PROC document is 41AL-1RK1C R031 this revision is later than the revision indicated by Risk Spectrum which is 41AL-1RK1C R030	Document Revision change only. No impact.
2004-150	The NIRM PROC document is 41AL-1RK5A R029 this revision is later than the revision indicated by Risk Spectrum which is 41AL-1RK5A R028	Document Revision change only. No impact.
2004-151	The NIRM PROC document is 41AL-1RK5B R023 this revision is later than the revision indicated by Risk Spectrum which is 41AL-1RK5B R022	Document Revision change only. No impact.
2004-152	The NIRM PROC document is 40OP-9WC01 R015 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9WC01 R014	Document Revision change only. No impact.
2004-153	The NIRM PROC document is 73ST-9SG01 R019 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9SG01 R018	Document Revision change only. No impact.
2004-154	The NIRM PROC document is 40AL-9RK3A R010 this revision is later than the revision indicated by Risk Spectrum which is 40AL-9RK3A R009	Document Revision change only. No impact.
2004-155	The NIRM PROC document is 40OP-9CH05 R007 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9CH05 R006	Documentation issue not expected to have a significant impact to CDF or LERF. In any event, the delta-CDFs calculated for this application would not be affected.
2004-156	The NIRM PROC document is 40DP-9OP19 R077 this revision is later than the revision indicated by Risk Spectrum which is 40DP-9OP19 R076	Document Revision change only. No impact.
2004-157	The NIRM DWG document is 01-E-NGA-0001 R004 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NGA-0001 R003	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-168	The NIRM PROC document is 40OP-9ZZ05 R096 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9ZZ05 R092	Document Revision change only. No impact.
2004-169	The NIRM PROC document is 14FT-1FP03 R007 this revision is later than the revision indicated by Risk Spectrum which is 14FT-1FP03 R006	Document Revision change only. No impact.
2004-170	The NIRM PROC document is 14FT-9FP28 R015 this revision is later than the revision indicated by Risk Spectrum which is 14FT-9FP28 R014	Document Revision change only. No impact.
2004-171	The NIRM PROC document is 40EP-9EO09 R019 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO09 R018	Document Revision change only. No impact.
2004-172	The NIRM PROC document is 40EP-9EO10 R033 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO10 R031	Document Revision change only. No impact.
2004-173	The NIRM PROC document is 41AL-1RK2A R043 this revision is later than the revision indicated by Risk Spectrum which is 41AL-1RK2A R042	Document Revision change only. No impact.
2004-174	The NIRM PROC document is 36ST-9SB14 R014 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SB14 R013	This impact is resolved. Documentation issue only. No impact.
2004-175	The NIRM PROC document is 40AL-9RK3A R011 this revision is later than the revision indicated by Risk Spectrum which is 40AL-9RK3A R009	Document Revision change only. No impact.
2004-176	The NIRM PROC document is 40AO-9ZZ12 R019 this revision is later than the revision indicated by Risk Spectrum which is 40AO-9ZZ12 R018	Document Revision change only. No impact.
2004-177	Add components 1MFPNHV0802 and 1MFPNHV0803 to the components table in Risk Spectrum. Link to same BEs as 1JFPNHV0802 and 1JFPNHV0803.	This impact is resolved. Documentation issue only. No impact.

ChangeID	Change_Description	Disposition
2004-187	The NIRM DWG document is 01-E-SGB-0019 R006 this revision is later than the revision indicated by Risk Spectrum which is 01-E-SGB-0019 R005	This addresses new blowdown valves installed with new steam generators. No measurable impact to CDF or LERF is expected.
2004-188	The NIRM PROC document is 40DP-9OP06 R074 this revision is later than the revision indicated by Risk Spectrum which is 40DP-9OP06 R071	Document Revision change only. No impact.
2004-189	The NIRM PROC document is 40OP-9ZZ04 R046 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9ZZ04 R042	Document Revision change only. No impact.
2004-190	The NIRM PROC document is 40OP-9SA01 R019 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9SA01 R017	Document Revision change only. No impact.
2004-191	The NIRM PROC document is 73ST-9XI33 R031 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI33 R029	Document Revision change only. No impact.
2004-192	The NIRM PROC document is 73ST-9XI09 R009 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI09 R008	Document Revision change only. No impact.
2004-193	The NIRM PROC document is 73ST-9XI10 R013 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI10 R012	Document Revision change only. No impact.
2004-194	The NIRM PROC document is 40DP-9OP29 R028 this revision is later than the revision indicated by Risk Spectrum which is 40DP-9OP29 R026	Document Revision change only. No impact.
2004-195	The NIRM PROC document is 40EP-9EO01 R011 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO01 R010	Document Revision change only. No impact.
2004-196	The NIRM PROC document is 40EP-9EO03 R016 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO03 R014	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-205	There are 10 check valves model in the AF System. Two (AFA-V005 and AFB-V009) are modeled for failure modes FO and RO. The remaining eight are only modeled with FO failure modes.	This impact is resolved. No model change was warranted.
2004-206	The NIRM CALC document is 01-EC-MA-0221 R009 this revision is later than the revision indicated by Risk Spectrum which is 01-EC-MA-0221 R008	Document Revision change only. No impact.
2004-207	The NIRM DBM document is IA R009 this revision is later than the revision indicated by Risk Spectrum which is IA R008	Document Revision change only. No impact.
2004-208	The NIRM DWG document is 01-E-NHA-0019 R013 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NHA-0019 R012	Document Revision change only. No impact.
2004-209	The NIRM DWG document is 01-M-SCP-0001 R048 this revision is later than the revision indicated by Risk Spectrum which is 01-M-SCP-0001 R046	Document Revision change only. No impact.
2004-210	The NIRM DWG document is 01-P-SGF-0120 R004 this revision is later than the revision indicated by Risk Spectrum which is 01-P-SGF-0120 R003	Document Revision change only. No impact.
2004-211	The NIRM DWG document is 01-P-RCF-0149 R001 this revision is later than the revision indicated by Risk Spectrum which is 01-P-RCF-0149 R000	Document Revision change only. No impact.
2004-212	The NIRM DWG document is A0-M-FPP-0005 R029 this revision is later than the revision indicated by Risk Spectrum which is A0-M-FPP-0005 R028	Document Revision change only. No impact.
2004-213	The NIRM DWG document is 01-M-WCP-0001 R025 this revision is later than the revision indicated by Risk Spectrum which is 01-M-WCP-0001 R023	Document Revision change only. No impact.
2004-214	The NIRM PROC document is 40DP-9OP06 R075 this revision is later than the revision indicated by Risk Spectrum which is 40DP-9OP06 R074	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-225	The NIRM PROC document is 30DP-9MT03 R009 this revision is later than the revision indicated by Risk Spectrum which is 30DP-9MT03 R008	Document Revision change only. No impact.
2004-226	The process of getting into the AF pump rooms has been changed. RE-AFA-LOCAL must now include these new steps contained in 40DP-9ZZ19.	Impact answered, ready for tech review. No change is expected to result. However, if a change does result, there would be no impact to the delta-CDFs calculated for this application.
2004-227	Add the unavailability events for the Condensate Pumps to the fault trees. They were inadvertently left out during the implementation of Impact 2002-110.	This impact is resolved. There is no impact to CDF or LERF.
2004-228	The NIRM DWG document is 01-M-SIP-0001 R029 this revision is later than the revision indicated by Risk Spectrum which is 01-M-SIP-0001 R026	Document Revision change only. No impact.
2004-229	The NIRM PROC document is 73ST-9XI20 R017 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI20 R016	Document Revision change only. No impact.
2004-23	Re-examine mission time for charging system. 24 hours was based on RCP seals (AssumptionCH013, which is no longer needed). According to SC_CH02, APSS has an actual MT requirement of only 8 hours. ATWS requires only 15 minutes (SC_CH01).	This is resolved. There was a slight decrease to CDF. There would be no impact to the delta-CDFs calculated for this application.
2004-230	The NIRM PROC document is 36ST-9SI04 R016 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SI04 R015	Document Revision change only. No impact.
2004-231	The NIRM PROC document is 36ST-9SI05 R014 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SI05 R013	Document Revision change only. No impact.
2004-232	The NIRM PROC document is 40ST-9SI04 R003 this revision is later than the revision indicated by Risk Spectrum which is 40ST-9SI04 R002	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-243	Delete parameters that are orphaned or generic parameters that have no basis events assigned consistent with 13-NS-B063 R6.	No impact.
2004-244	Reassign existing Basic Events to new parameters generated by 13-NS-B063 R6.	No impact.
2004-245	Revise existing RS parameters to new values generated by 13-NS-B063 R6.	Editorial changes only. No impact.
2004-246	The NIRM CALC document is 13-MC-SI-0309 R004 this revision is later than the revision indicated by Risk Spectrum which is 13-MC-SI-0309 R003	Document Revision change only. No impact.
2004-247	The NIRM DWG document is 01-E-NKA-0001 R009 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NKA-0001 R008	Document Revision change only. No impact.
2004-248	The NIRM DWG document is 01-E-NKA-0002 R006 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NKA-0002 R005	Document Revision change only. No impact.
2004-249	The NIRM DWG document is 01-E-PKA-0002 R016 this revision is later than the revision indicated by Risk Spectrum which is 01-E-PKA-0002 R015	Document Revision change only. No impact.
2004-25	Update unfavorable MTC parameters for current core construction, which does not go less negative than -0.61 and goes less negative for longer for -0.77.	The impact of this change is a 1.2% increase in internal CDF and 2.2% increase in internal LERF, therefore the priority is low. There would be no impact to the delta-CDFs calculated for this application.
2004-250	The NIRM DWG document is 01-E-PKA-0005 R009 this revision is later than the revision indicated by Risk Spectrum which is 01-E-PKA-0005 R008	Document Revision change only. No impact.
2004-251	The NIRM DWG document is 01-E-NHA-0008 R008 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NHA-0008 R007	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-262	The NIRM PROC document is 40EP-9EO10 R035 this revision is later than the revision indicated by Risk Spectrum which is 40EP-9EO10 R034	Document Revision change only. No impact.
2004-263	The NIRM PROC document is 73DP-9XI01 R015 this revision is later than the revision indicated by Risk Spectrum which is 73DP-9XI01 R009	Document Revision change only. No impact.
2004-264	The NIRM PROC document is 55OP-0GT01 R040 this revision is later than the revision indicated by Risk Spectrum which is 55OP-0GT01 R039	Document Revision change only. No impact.
2004-265	The NIRM PROC document is 40OP-9CH01 R037 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9CH01 R036	Document Revision change only. No impact.
2004-266	The NIRM PROC document is 40DP-9OP19 R082 this revision is later than the revision indicated by Risk Spectrum which is 40DP-9OP19 R080	Document Revision change only. No impact.
2004-267	In the formula.txt files for units 1-3, there are errors and omissions in the IEATWS2 equation. IEATWS2 equation should match the IETT equation.	EOOS issue.
2004-268	Add component to system links that were neglected in the resolution of Impact 2004-180.	EOOS issue.
2004-269	Impact 2004-132 did not specify EOOS changes (S/U Transformer SWYD breakers OOS (both) incorrectly fails power to NAN-S05/6 even when loads are transferred to the Alternate S/U Transformer)	EOOS issue.
2004-270	In the EOOS_Test_Table_generator.mdb, in the associated TEST tables, the train designator for 72PA-9ZZ08 is missing. Also in the EOOS.mdb, in the TEST_U3, 72PA-9ZZ08 has the wrong unit designator.	EOOS issue.

ChangeID	Change_Description	Disposition
2004-281	The NIRM PROC document is 40OP-9ZZ04 R047 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9ZZ04 R046	Document Revision change only. No impact.
2004-282	The NIRM PROC document is 40ST-9DG02 R025 this revision is later than the revision indicated by Risk Spectrum which is 40ST-9DG02 R024	Document Revision change only. No impact.
2004-283	The NIRM PROC document is 40ST-9DG01 R022 this revision is later than the revision indicated by Risk Spectrum which is 40ST-9DG01 R021	Document Revision change only. No impact.
2004-284	The NIRM PROC document is 32MT-9ZZ58 R022 this revision is later than the revision indicated by Risk Spectrum which is 32MT-9ZZ58 R021	Document Revision change only. No impact.
2004-285	The NIRM PROC document is 73ST-9XI33 R032 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI33 R031	Document Revision change only. No impact.
2004-286	The NIRM PROC document is 36ST-9SI04 R017 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SI04 R016	Document Revision change only. No impact.
2004-287	The NIRM PROC document is 40AO-9ZZ07 R014 this revision is later than the revision indicated by Risk Spectrum which is 40AO-9ZZ07 R013	Document Revision change only. No impact.
2004-288	The NIRM PROC document is 36ST-9SA01 R030 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SA01 R029	Document Revision change only. No impact.
2004-289	The NIRM PROC document is 73ST-9SI06 R017 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9SI06 R016	Document Revision change only. No impact.
2004-290	The NIRM PROC document is 40OP-9SI02 R054 this revision is later than the revision indicated by Risk Spectrum which is 40OP-9SI02 R053	Document Revision change only. No impact.

ChangeID	Change_Description	Disposition
2004-301	The NIRM PROC document is 40AO-9ZZ12 R020 this revision is later than the revision indicated by Risk Spectrum which is 40AO-9ZZ12 R019	Document Revision change only. No impact.
2004-302	Loss of Turbine Cooling Water is modeled for Instrument Air Compressors, but loss of cooling due to blockage in the Air Compresssor After Cooler is not modeled and has no Memo Assumption explaning why this should not be modeled.	Document issue. No impact.
2004-57	Determine the appropriate RCS T_ave for units 1, 3. Incorporate that T_ave into 13-NS-C036 (into PVA.par, parameter TWPSNM & TWPSQ). And determine the impact on MAAP4.0.4 existing applications.	This is not expected to have any measurable impact to CDF or LERF. There would be no impact to the delta-CDFs calculated for this application.
2004-59	The DLTRM fault tree contains events that have been identified as #EOOS# events. If these events are set to true then the DLTRM tree solution removes valid cutsets as well as the invalid ones. Fix the tree.	EOOS issue only. No impact.
2004-72	Address discrepancy between how local valve failure and control circuit failure are modeled for the SI pump combined miniflow valves, SIA-UV659 and SIB-UV660; one is tested, the other is mission time. Ref gates GLI137 and GLI237.	This is resolved. Less than 1% change to CDF and LERF. There would be no impact to delta-CDFs calculated for this application.
2004-73	The ATWS4 event tree has sequences that are going to a Level 2 PKA tree but the boundary condition set for the ATWS tree is MISC.	The ATWS4 tree was run using first the MISC BC set, then the PKAM41 BC set. The PKA was only slightly higher. No significant impact to CDF. There would be no impact to delta-CDFs calculated for this application.
2004-75	Correct application of error factor for ten probability parameters to bring the 95th percentile value to <=1.00. Uncertainty analysis must have probability values less than or equal to 1.0.	Mean values are not affected, so these is no impact to the results.
2004-76	Consider PSV relief and induced-LOCA in SBO tree following SGHR-E success. MAAP indicates PSVs begin lifting at about 20 Minutes. A large part of AFW-A failure is recoverable manually, but time to effect may be longer than 20 minutes.	Documentation only. No impact.

ChangeID	Change_Description	Disposition
2005-4	The NIRM DWG document is 01-E-NAB-0021 R003 this revision is later than the revision indicated by Risk Spectrum which is 01-E-NAB-0021 R002	Document Revision change only. No impact.
2005-5	The NIRM DWG document is A0-E-NAA-0006 R002 this revision is later than the revision indicated by Risk Spectrum which is A0-E-NAA-0006 R001	Document Revision change only. No impact.
2005-6	The NIRM PROC document is 36ST-9SA02 R029 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SA02 R028	Document Revision change only. No impact.
2005-7	The NIRM PROC document is 73ST-9XI24 R006 this revision is later than the revision indicated by Risk Spectrum which is 73ST-9XI24 R005	Document Revision change only. No impact.
2005-8	The NIRM PROC document is 40ST-9SI04 R004 this revision is later than the revision indicated by Risk Spectrum which is 40ST-9SI04 R003	Document Revision change only. No impact.
2005-9	The NIRM PROC document is 36ST-9SA01 R031 this revision is later than the revision indicated by Risk Spectrum which is 36ST-9SA01 R030	Document Revision change only. No impact.