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
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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

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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

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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. [

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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). [

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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.) [

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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.

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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.

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3.1.3.2 Post-LOCA Cooldown Methodology

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3.1.3.3 Post-RAS HPSI Status

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3.1.3.4 Post-RAS CSS Status

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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

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4.2.1 Detailed Profile of the 3-Inch Case

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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:

[ ]

[ ]

[ ]

[redacted]

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]

[ ]

[ ]

[ ]

[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.

[ ]

[ ]

[ ]

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

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