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AP1000 Code Applicability Report

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LIST OF ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safety
ADS	Automatic Depressurization system
APEX	Advanced Plant Experiment
CCFL	Counter Current Flow Limitation
CCTF	Cylindrical Core Test Facility
CFD	Computational Fluid Dynamics
CHF	Core Heat Transfer
CMT	Core Makeup Tank
COSI	Condensation of Safety Injection
CQD	Code Qualification Document
CSAU	Code Scaling, Applicability, and Uncertainty
DBA	Design Basis Accident
DCD	Design Control Document
DECLB	Double-Ended Cold Leg Break
DEG	Double-Ended Guillotine
DEDVI	Double-Ended Direct Vessel Injection
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EPRI	Electric Power Research Institute
FLECHT-SEASET	Full-length Emergency Cooling Heat Transfer – Systems Effects and Separate Effects Test
FSER	Final Safety Evaluation Report
GDC	General Design Criteria
GE	General Electric
HDR	Heissdampfreaktor
HX	Heat Exchanger
IRWST	In-Containment Refueling Water Storage Tank
LBLOCA	Large Break LOCA
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
LST	Large Scale Test
LSTF	Large Scale test Facility (ROSA IV)
MSLB	Main Steam Line Break
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulation
ORNL	Oak Ridge National Laboratory
OSU	Oregon State University
PCCWST	Passive Containment Cooling System Water Storage Tank
PCS	Passive Containment Cooling System
PCT	Peak Clad Temperature

LIST OF ACRONYMS AND ABBREVIATIONS (cont.)

PIRT	Phenomena Identification and Ranking Table
PRHR	Passive Residual Heat Removal
PWR	Pressurized Water Reactor
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
ROSA	Rig of Safety Assessment
RTDP	Revised Thermal Design Procedure
SBLOCA	Small-break LOCA
SDSER	Supplemental Draft Safety Evaluation Report
SG	Steam Generator
SGTR	Steam Generator Tube Rupture
SI	Safety Injection
SIMARC	Simulator Advanced Real-time Code
SLB	Steam Line Break
SPES	Simulatore per Esperienze di Sicurezza
THTF	Thermal Hydraulic Test Facility
TMI	Three-Mile Island
UPTF	Upper Plenum Test Facility
V&V	Verification and Validation

EXECUTIVE SUMMARY

This report documents an assessment performed by Westinghouse of the analysis codes that were developed and approved for the AP600 Design Certification to determine their applicability and use for Design Certification of an AP1000. The analysis codes that were approved for the purposes of performing safety analyses of the AP600 passive plant are:

- LOFTRAN – transient analyses
- NOTRUMP – small-break LOCA analyses
- WCOBRA/TRAC – large break LOCA & long-term cooling analyses
- WGOTHIC – containment analyses

The report describes the basis for the use of these safety analysis codes approved for the AP600, a plant design with passive safety features, for a Design Certification of an AP1000. For each of the thermal-hydraulic analysis codes, the report discusses the basis for that approval as described in NUREG-1512, Final Safety Evaluation Report Related to Certification of the AP600 Standard Design (Reference 1). This report also provides an assessment as to how that basis can be applied to AP1000.

Background

As part of the pre-certification review of the AP1000, Westinghouse submitted WCAP-15612, "AP1000 Plant Description and Analysis Report" (Reference 2) to the NRC. That report provides an overview description of the AP1000 plant, and compares its important design features to those of the AP600. The AP1000, which is based on the AP600 design, has the same plant footprint as the AP600. In addition, the configuration and operation of the reactor coolant system and the passive safety features are the same. Components and pipe dimensions have been increased, where needed, to accommodate the higher core power of the AP1000, but the basic configuration (i.e. number of components and how they are interconnected) is the same. In Reference 2, analyses of representative design basis accidents for the AP1000 and are compared to the results from the AP600 safety analyses. These analyses were performed using the codes and methods that were utilized and approved for the AP600. These analyses do not represent the complete spectrum of design basis accidents for AP1000. Rather, they represent a sampling of the design basis accidents where the performance of the passive safety systems is critical in mitigating the consequences of the accident. These assessments are, therefore, useful in characterizing the performance of and assessing the phenomena associated with the AP1000 passive safety systems. Results of these analyses show similar behavior for both the AP600 and AP1000.

Westinghouse has submitted WCAP-15613, "AP1000 PIRT and Scaling Assessment" (Reference 3) to the NRC. The report provides an assessment of the AP600 test program and its applicability to AP1000 and provides Phenomenon Importance Ranking Tables (PIRT), which were developed for the AP1000 based on an independent review performed by several industry experts. The report addresses the applicability of each of the AP600 test facilities that were important for Design Certification. The important separate effect tests and integral effects tests were evaluated in more detail, to demonstrate that the test data and conclusions obtained from these tests are applicable to AP1000. In-depth scaling analyses of the AP600 integral effects tests such as OSU and SPES-2 were performed to demonstrate that the integral effect tests are adequately scaled for AP1000. The major conclusion from this report is that the AP600 test program can be judged to meet the requirements of 10 CFR Part 52 for an application for Design Certification for

AP1000. More specifically, the tests provide an adequate database to validate analysis codes for the purposes of performing safety analysis for an AP1000. Analysis codes that are validated against this test data can be used to perform the required accident analyses for AP1000.

Scope of this Report

In this report, Westinghouse describes the bases for the use of the analysis codes previously validated and approved for AP600 for Design Certification of the AP1000. For each of the thermal-hydraulic analysis codes that were developed and approved as part of AP600 Design Certification, (LOFTRAN, NOTRUMP, WCOBRA/TRAC, and WGOTHIC), the report discusses the basis for that approval, as described in Reference 1. A summary of the major issues for each code is provided with a discussion of the applicability of the AP600 code approval basis to the AP1000. This provides the justification for the continued use of these codes for AP1000.

The following summarizes the conclusions of this report:

- The LOFTRAN-AP code that was approved for AP600 can be used for the purposes of performing conservative analyses of the transient events presented in Chapter 15 for AP1000. The basis for this conclusion is that when considering transient events, no new phenomena are identified for AP1000, when compared to AP600, and the test database that supported validation of this code for AP600 is applicable to AP1000. Furthermore, the means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000. Assessments indicate that the AP1000 passive safety systems operate in the same way as the AP600, and that large margins to the regulatory limits exist for the transient events analyzed. Large margins exist for the AP1000 Chapter 15 accident analysis events analyzed with LOFTRAN.

- The NOTRUMP code that was approved for AP600 can be used for the purposes of performing conservative (Appendix K) analyses of the small break LOCA events presented in Chapter 15 for AP1000. For small break LOCA events, no new phenomena are identified for AP1000, when compared to AP600, and the test database that supported validation of this code for AP600 is applicable to AP1000. Also, the means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000. Large margins exist for the Chapter 15 accident analysis events analyzed with NOTRUMP. It was noted in Reference 3 that some phenomena previously addressed for AP600 could be judged to be of higher importance for AP1000 (i.e., entrainment in the hot leg during the transition from ADS to IRWST injection of the SBLOCA event). To better address these phenomena, additional justification is provided as follows:

  - Sensitivity studies show that AP1000 SBLOCA performance is relatively insensitive to hot leg/upper plenum entrainment and that acceptable core cooling is maintained even when higher than expected entrainment (homogenous flow assumed in upper plenum, hot legs, and ADS-4) is assumed (Appendix F).

- Comparison of the NOTRUMP level swell model to full-scale bundle data confirms the validation of this aspect of NOTRUMP (Appendix G).

- Comparison of NOTRUMP predications to integral systems test data specific to AP1000 provide additional validation of NOTRUMP for AP1000 (Appendix E).

- The WCOBRA/TRAC code that was approved for AP600 large break LOCA analysis can be used for the purposes of performing best-estimate analysis for AP1000. The basis for this conclusion is that for large break LOCA events, no new phenomena are identified for AP1000, when compared to AP600, and the test database that supported validation of this code is applicable to AP1000. Furthermore, the means of resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000. The additional validation of WCOBRA/TRAC to address the uniqueness of the passive safety system direct vessel injection (DVI) has been performed and approved by the NRC for AP600. As the AP1000 DVI is the same as AP600, this validation is applicable to AP1000 as well. The WCOBRA/TRAC computer code and large break LOCA methodology approved by the NRC for AP600 are applicable to the 10CFR50.46 Emergency Core Cooling System performance analysis of the AP1000 for 95th percentile calculated peak clad temperature values up to the 2200°F licensing limit.

- The WCOBRA/TRAC code that was approved for AP600 long-term cooling analysis can be used for the purposes of performing conservative (Appendix K) analysis of long-term cooling for LOCA events presented in Chapter 15 for AP1000. The basis for this conclusion is that for LOCA events, no new phenomena are identified for AP1000, when compared to AP600, and the test database that supported validation of this code for AP600 is applicable to AP1000. Also, the means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000. The AP1000 passive safety systems provided large margins to the regulatory limits for the accident analysis events analyzed with WCOBRA/TRAC for long-term cooling. Note however that in Reference 1, the use of WCOBRA/TRAC for long-term cooling in the "window" mode (as approved for AP600) was compared to an analysis using a "continuous" mode for the limiting long-term cooling event. Results of that analysis demonstrated good agreement between the window mode analysis and the continuous mode analysis. Westinghouse has performed the limiting long-term cooling analysis using the continuous mode methodology presented in Reference 1, but retains the windows mode methodology for the less limiting events to minimize the resources expended to perform this analysis. Comparison of the results of the continuous mode to the window mode supports the assessment of conservative results for the "window" mode analyses.

- The WGOTHIC code that was approved for AP600 can be used for the purposes of performing conservative containment analysis of the events presented in Chapter 6 for AP1000. The basis for this conclusion is that regarding the events that challenge containment integrity (i.e., large LOCA and large steam line break), no new phenomena are identified for AP1000, when compared to AP600, and the test database that supported validation of this code for AP600 is applicable to AP1000. Furthermore, the means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000. The AP1000 has sufficient margin to the containment design pressure when bounding-type analyses are performed using WGOTHIC.

Conclusion

The analysis codes were extensively reviewed by the NRC as part of the AP600 Design Certification process. The review conducted by the staff included key elements of Draft Regulatory Guide DG-1096. There are no new phenomena associated with the AP1000, and scaling demonstrates that the AP600 test database used to validate the analysis codes is applicable to AP1000. Similar plant margins exist between AP600 and AP1000. Therefore, the analysis codes are acceptable for use on AP1000.

References

1. NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998.

2. WCAP-15612, "AP1000 Plant Description and Analysis Report," December 2000.

3. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

1.0 INTRODUCTION

Westinghouse Electric Company has designed an advanced 600 MWe nuclear power plant called the AP600. The AP600 uses passive safety systems to enhance plant safety and to satisfy U.S. licensing requirements. The use of passive safety systems provides significant and measurable improvements in plant simplification, safety, reliability, investment protection, and plant costs. These systems use only natural forces such as gravity, natural circulation, and compressed gas to provide the driving forces for the systems to adequately cool the reactor core following an accident. The AP600 received Design Certification by the Nuclear Regulatory Commission in December 1999.

To further improve AP600 economics and in response to market demand for larger plants, Westinghouse initiated development of the AP1000 standard nuclear reactor design, with an output of approximately 1000 MWe, based upon the AP600 design. The design features of the plant have been selected to preserve key features and performance characteristics embodied in the AP600. By preserving the design basis of the AP600 in the AP1000, Westinghouse seeks to preserve the licensing basis of the plant as well.

Westinghouse submitted the "AP1000 Plant Analysis and Description Report" (Reference 1) to the NRC. The report provides a description of the AP1000 plant design as well as accident analyses using the AP600 validated analysis codes and preliminary models of the AP1000 plant. These preliminary safety analyses are not a complete set of analyses as prescribed by 10CFR Part 50, but rather, were provided to characterize the expected performance of the AP1000.

Westinghouse submitted the "AP1000 PIRT and Scaling Assessment" (Reference 2) report to the NRC. The report provides Phenomena Identification and Ranking Tables (PIRT) for the AP1000 and demonstrates through scaling that the AP600 test program is applicable to the AP1000 and sufficiently covers the range of conditions expected for the AP1000. The report concludes that the AP600 test program provides a test database sufficient for code validation for AP1000 in accordance with 10CFR Part 52.

This report documents the acceptability of the analysis codes approved for AP600 for application to AP1000. The basis for approval for AP600 is discussed along with major code-related issues identified during the AP600 Design Certification review, and the means to address these issues as the codes are applied to the AP1000 are presented. Each section provides an assessment of how the AP600 code approval basis can be applied to AP1000.

Section 2 addresses acceptance of the WCOBRA/TRAC code for AP600 large break LOCA and long-term cooling analysis. It also addresses the acceptability of the WCOBRA/TRAC code for AP1000 large break LOCA and long-term cooling analysis. Sections 3, 4, and 5 address the acceptability of the NOTRUMP, LOFTRAN, and WGOTHIC codes, respectively, for use in analyzing AP1000. Section 6 provides conclusions regarding the applicability of the AP600 analysis codes to AP1000.

References

1. WCAP-15612, "AP1000 Plant Description and Analysis Report," December 2000.

2. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

2.0 WCOBRA/TRAC COMPUTER CODE VALIDATION FOR AP1000

2.1 AP1000 LARGE BREAK LOCA PHENOMENA

Table 2-1 shows key processes for the large break LOCA (LBLOCA) transient. The LBLOCA transients include double-ended guillotine (DEG) breaks, and large cold-leg split breaks with flow area greater than 1 ft^2. These transients are initiated at full-power conditions with the plant parameters either at best estimate values or bounded in a conservative manner. The uniqueness of the AP1000 plant is assessed relative to AP600 and to existing PWRs to identify any differences in the plant design that could affect WCOBRA/TRAC's capabilities for modeling the AP1000.

The assessment of safety analysis code capability for the AP1000 LBLOCA analysis is performed for WCOBRA/TRAC, the code that will be used to perform the analysis. The bases for using WCOBRA/TRAC are:

- It is the highest level of thermal-hydraulic technology among the industry LOCA analysis licensing codes. It has the most complete thermal-hydraulics model for analyzing the complex behaviors associated with large LOCA events.

- It has already been reviewed by the NRC and approved as a best-estimate code consistent with Regulatory Guide 1-157, "Best-Estimate Calculations of ECCS Performance" (Reference 1). Westinghouse and EPRI developed this best-estimate LOCA methodology under the revision to the Appendix K rule (1988), and it has been used in more than ten Westinghouse four-loop and three-loop plant LBLOCA licensing analyses to date to calculate the peak cladding temperature at the 95th percentile.

- A code qualification document (CQD) (Reference 2) exists for WCOBRA/TRAC, and a nodalization scheme of the AP600 design was approved in WCAP-14601 (Reference 3). WCOBRA/TRAC has also been validated against experiments that capture the key LBLOCA processes for the AP600. Section 2.3 of WCAP-15613, "AP1000 PIRT and Scaling Assessment" (Reference 4) presents the AP1000 LBLOCA PIRT and concludes that the new and additional passive systems do not significantly influence the LBLOCA calculated peak clad temperature (PCT). Only the downcomer injection requires specific additional validation of WCOBRA/TRAC for passive plants.

- This additional validation of WCOBRA/TRAC to address the uniqueness of the passive safety system direct vessel injection (DVI) has been performed in WCAP-14171, Rev. 2 (Reference 5) and approved by the NRC for AP600. This validation, which also applies to the AP1000, used test data that exist on DVI from the full-scale upper plenum test facility (UPTF) (Reference 6) tests, part of the NRC cooperative program with the Federal Republic of Germany, and the Japanese cylindrical core test facility (CCTF) (Reference 7) reflood system effects tests that model a four-loop Westinghouse PWR with the DVI configuration. Further, sufficient data existed for the DVI configuration that no specific AP600 test was needed to provide data to validate WCOBRA/TRAC for injection into the reactor vessel downcomer. Section 2.3 of Reference 4 concludes that the downcomer injection location validation of WCOBRA/TRAC performed for AP600 addresses the issue for AP1000.

Examples of the WCOBRA/TRAC validation documented in the CQD are provided in Table 2-2.

Although a very small effect, the core makeup tanks (CMT) actuate during a LBLOCA event before the accumulators inject. The amount of CMT injection that occurred in the AP600 LBLOCA analysis is small (Figure 2-1). Only about 0.5 percent of the CMT liquid inventory was injected before accumulator flow shut off the CMT injection for AP600, and this water did not contribute to core cooling because it bypassed the reactor vessel. A similar result is anticipated for AP1000 because its passive safety system design is very similar to AP600.

A CMT test was performed to provide thermal-hydraulic data that covered the expected range of conditions for the AP600. WCOBRA/TRAC modeled these experiments in order to validate the correlations in the code used for condensation. There is very little CMT injection during a LBLOCA because the rapid depressurization causes the accumulator flow to begin early in the transient, shutting off CMT injection. As a result, the core recovered and the peak cladding temperature excursion is terminated via accumulator flow, not CMT flow. The same result is anticipated for AP1000 large break LOCA analyses. The AP1000 Passive Residual Heat Removal (PRHR) is actuated during a large LOCA event but has little impact because of the massive depressurization that occurs due to the postulated double-ended cold leg break (DECLB).

The PIRT review of the key LBLOCA phenomena presented in Reference 4 indicates that, as is true for AP600, the unique passive safety systems of AP1000 play almost no role in the plant's response during the PCT excursion of a LBLOCA event because the transient is so rapid. Westinghouse evaluated the need for performing a LBLOCA test and considered it to be unnecessary for AP600; the same conclusion holds true for AP1000. Furthermore, the AP1000 design features that are no different from conventional Westinghouse plants require no testing. Data for computer code validation exist for the phenomena associated with DVI during the AP1000 LBLOCA transient.

The long-term cooling aspects of the LBLOCA are the same as for the small break LOCA (SBLOCA) that were studied at the Oregon State University (OSU) test facility. Long-term cooling will be analyzed as an event separate from the initiating event, as discussed in Subsection 2.3.

2.2 WCOBRA/TRAC CODE VALIDATION FOR AP1000 LBLOCA ANALYSIS

2.2.1 WCOBRA/TRAC Acceptance for AP600 LBLOCA Analysis

In Section 21 of the AP600 FSER (Reference 8), the NRC staff reported the results of its review of the Westinghouse LBLOCA methodology submittal. The staff concurred that the extensive assessment of the WCOBRA/TRAC computer code performed for conventional three-loop and four-loop plant LBLOCA analysis applied to AP600 because of the similarity of the transient responses. Based on their previous review of the three-loop and four-loop plant large break LOCA methodology, together with the AP600-related validation and assessments provided in WCAP-14171 (Reference 5), the staff further concluded that the WCOBRA/TRAC code is "adequate to provide realistic evaluations of the AP600 LBLOCA with the tendency toward conservative results."

The Phenomena Identification and Ranking Table (PIRT) performed for the AP600 large break LOCA in WCAP-14171 and the corresponding AP1000 PIRT in Reference 4 indicate that the new, additional

passive safety systems did not influence the calculated peak clad temperature; the one issue that was identified in the AP600 PIRT is the DVI configuration. The effect of the downcomer injection location was addressed by specific additional validation. First, the Japanese CCTF DVI test number 58 has been simulated with WCOBRA/TRAC. The CCTF test facility is a full-height, lower pressure model of a four-loop Westinghouse PWR. The scale factor for the facility compared to a four-loop plant is about 1/20. The facility was specifically designed to investigate the gravity reflood systems behaviors following a LBLOCA. Test 58 simulates the reflood portion of the large-break transient, during which the accumulator and low-pressure pumped flow is injected into the downcomer of the test vessel to quench the heated core. The test models the heated core with full-length heater rods, the reactor vessel, steam generators, and associated piping. The DVI configuration is not exactly the same as the AP600 or AP1000, since there is no flow-turning device in the CCTF downcomer simulation. As a result, the injected flow will spread more in this test facility than in either advanced passive plant.

Figure 2-2 shows the CCTF facility, and Figure 2-3 shows the facility downcomer and the injection locations. Modeling this test with WCOBRA/TRAC has verified the ability of the interfacial heat and mass transfer models used in the downcomer to calculate the amount of condensation that occurs during accumulator injection and safety injection with the DVI configuration.

To address the issue of the effects of DVI on emergency core cooling (ECC) bypass during the AP600 LBLOCA event, the UPTF experiment with DVI was also modeled with WCOBRA/TRAC. The UPTF facility was constructed to investigate the LBLOCA ECC bypass phenomena. The UPTF uses a full-scale, four-loop reactor vessel and downcomer. Experiments were conducted with DVI using the accumulator and the pumped flows of LBLOCA refill conditions. Figure 2-4 shows the UPTF, and Figure 2-5 shows the UPTF test vessel. Prediction of this test also confirmed the interfacial heat and mass transfer models used in the WCOBRA/TRAC code. The NRC stated in Reference 8, the AP600 FSER, that the WCOBRA/TRAC computer code realistically predicts the DVI test configuration data from the CCTF and UPTF facilities in WCAP-14171.

Additional validation was also performed to ensure that the WCOBRA/TRAC models and correlations apply over the extended ranges of blowdown cooling and reflood cooling conditions exhibited by the AP600 design. The results of ORNL test and FLECHT-SEASET test simulations presented in WCAP-14171 resolved any questions relating to the range of parameter validation that existed for the WCOBRA/TRAC large break LOCA heat transfer predictions of the AP600. Elements of the three-loop and four-loop plant best estimate LOCA methodology approved for Westinghouse plants were not performed for AP600 because the calculated PCT at the 95th percentile was below 1700°F in the AP600 SSAR (Reference 9) analysis.

The AP600 LBLOCA methodology was found to be acceptable relative to 10CFR50.46 and to the Regulatory Guide 1.157 guidance, subject to certain methodology and application restrictions. The large majority of these application restrictions are the same as those identified in the acceptance of the WCOBRA/TRAC large break LOCA methodology for three-loop and four-loop Westinghouse plant designs and are not repeated. The AP600-related restrictions in Section 21 of Reference 8 that deal with a reanalysis situation are discussed in the following subsection.

2.2.2 WCOBRA/TRAC Acceptability for AP1000 LBLOCA Analysis

As discussed in Section 2.2.1, WCOBRA/TRAC is the licensing code used for the LBLOCA analysis of the AP600. Table 2-1 indicates that, for a LBLOCA, AP1000 thermal-hydraulic performance is very similar to existing Westinghouse PWRs, with the exception of DVI. As discussed in Section 2.2.1, WCOBRA/TRAC was validated for predicting DVI phenomena in WCAP-14171; it had already been validated against ample data, on different scales, for the other thermal-hydraulic phenomena associated with a LBLOCA, as documented in Reference 2, the WCOBRA/TRAC Code Qualification Document.

As previously stated, the PIRTs for the AP600 and AP1000 LBLOCA events are almost identical. There are no additional phenomena that require any further validation or assessment of WCOBRA/TRAC for AP1000 LBLOCA analysis, so no novel features are needed in WCOBRA/TRAC and the LBLOCA model accepted for AP600 is acceptable for AP1000. The code will be applied as described below.

Code Version

A special version of the WCOBRA/TRAC computer code was created for the AP600 SSAR analysis by incorporating additional capability to model the unique features of the AP600, as documented in WCAP-14776 (Reference 10), Section 4. A similar approach is used to perform the AP1000 large break LOCA design certification analysis. The same updates added to WCOBRA/TRAC for the AP600 analysis are used in the creation of an "AP" version to perform the AP1000 large break LOCA licensing analysis. The "AP" version of WCOBRA/TRAC includes the discretionary and non-discretionary code changes that have been made since the AP600 SSAR analysis was performed, which constitute the "2000 formulation" of the code and which have been reported to the NRC by Westinghouse (Reference 11) per the 10CFR50.46 annual reporting process. The details of the code changes made since the AP600 analysis was performed are provided in Appendix A. The impact of implementing these changes into WCOBRA/TRAC is judged to be minor on the AP1000 large break LOCA results.

AP600 FSER Restrictions

The AP600 FSER (Reference 8) identified several items as restrictions on further AP600 WCOBRA/TRAC LBLOCA analyses, in the event that the 95th percentile PCT values for either blowdown or reflood exceeded 1725°F. The 95th percentile PCT for AP1000 will exceed this value. The NRC-specified requirements follow, together with the means by which the AP1000 analysis will comply with each:

1. Westinghouse shall "repeat the global model matrix of calculations and the final 95 percent uncertainty calculations." The reference transient and the global model matrix of cases will be executed in the AP1000 LBLOCA analysis in order to establish the final 95th percentile PCT value using the same uncertainty methodology as AP600.

2. Westinghouse shall "address the sensitivity to the CMT and PRHR modeling parameters...as a bias to the 95 percent PCT result." An AP1000 WCOBRA/TRAC case will be run in which the CMT is not modeled and another, separate case will be run in which the PRHR is not modeled. If either case produces a higher PCT than the base case, the PCT difference will be applied as a bias

in determining the final 95th percentile PCT value. Individual biases will be applied to the blowdown and reflood phase PCT results.

3. Westinghouse shall perform both local and core-wide oxidation calculations using the techniques approved for three-loop and four-loop plants. The oxidation calculation will be performed using the methods approved for use in three- and four-loop plant applications, as stipulated in the AP600 FSER, Section 2.1.6.3.

The AP1000 design certification large break LOCA analysis will conform to the identified restrictions. The methodology for determining the operation involves core heatup calculations and is independent of the passive plant design.

Major Issues

Inasmuch as the major issues identified during the AP600 review were resolved successfully in the AP600 design certification, and the AP600 approval is grounded in the generic PWR test database rather than AP600-specific testing, there are no major issues associated with the AP1000 large break LOCA analysis approach and/or phenomena. The WCOBRA/TRAC computer code and the large break LOCA best estimate methodology approved by the staff for AP600 are applicable to AP1000 for 95th percentile calculated PCT values up to the 2200°F licensing limit.

Resolution of Issues

The AP1000 LBLOCA Emergency Core Cooling System (ECCS) performance analysis will comply with the AP600 FSER restrictions, as indicated above. The nodalization used for the AP600 LBLOCA analysis in WCOBRA/TRAC will be adjusted to model the 14-foot core length of AP1000.

Conclusions

The calculated PCT for the AP1000 large break LOCA event will exceed the AP600 result because of the increase in core power. However, there are no new phenomena involved, and the AP1000 passive safety systems (other than accumulators) do not significantly impact the PCT for large break LOCA. The large break LOCA methodology used in the AP600 SSAR, including use of the WCOBRA/TRAC code version described above, is directly applicable to the 10CFR50.46 ECCS performance analysis of the AP1000 design.

2.3 WCOBRA/TRAC VALIDATION FOR AP1000 LONG-TERM COOLING ANALYSIS

2.3.1 Long-Term Cooling Phenomena

The AP1000 long-term core cooling process is different from that of conventional PWRs; under design basis safety analysis assumptions, there are no recirculating pumps to provide flow to the reactor vessel to maintain core cooling for post-accident situations. The AP1000 uses gravity-driven flow from the In-containment Refueling Water Storage Tank (IRWST) for the initial period of long-term cooling. Later, when the containment sump has filled with water, the containment recirculation phase begins.

Containment recirculation provides decay heat removal for days and weeks following a LOCA event, with energy being removed through the containment shell to the air and water of the containment cooling system. During the containment recirculation phase, ECCS water flows again by gravity into the reactor vessel through the DVI lines. In post-LOCA long-term cooling, gravity-driven phenomena dominate, and the processes are simple for any size break.

The long-term cooling phase of AP1000 LOCA events continues to be defined as it is for AP600 in Section 1 of WCAP-14776 (Reference 10). The long-term cooling processes are shown in Table 2-3. The plant configuration during this post-accident phase is characterized by the reactor vessel being partially filled, the vessel volume either in boiling or in single-phase convective flow, the core covered by either a two-phase or single-phase mixture, and the downcomer containing subcooled water. The primary system above the reactor hot legs has drained, and the main vent path out of the primary system is through the fourth stage ADS valves on the hot legs. The fourth-stage ADS valves are above the flood-up level of the sump. The IRWST and/or containment sump will inject flow into the reactor vessel once the isolation valves open and vessel pressure is lower than the driving head available. The reactor primary system and containment taken together form a closed natural circulation system in which the steam generated in the core is vented through the ADS and condensed on the containment shell, fed to the IRWST and/or containment sump as condensate, then injected into the reactor vessel downcomer.

WCOBRA/TRAC is the computer code used to model this post-accident period. The code's important modeling features are the ability to simulate multiple break points in the RCS and to preserve correct elevation heads in the natural circulation process. WCOBRA/TRAC is also accurate at low pressure and has been compared to several reflood system effects tests in the CQD (Reference 2), that have thermal-hydraulic characteristics similar to the post-LOCA accident phase for the AP1000.

As shown in Table 2-3, data existed for several but not all phenomena, prior to the AP600 Oregon State University experiments that examined the gravity-driven long-term cooling behavior of a passive safety system design similar to the AP1000. Reference 4, Section 2.4 concludes that there are no new long-term cooling phenomena for AP1000 relative to AP600. It further concludes that the AP600 test facilities are adequately scaled for AP1000. Therefore, no specific data are needed on the AP1000 long-term cooling phenomena beyond that identified in Table 2-3.

2.3.2 WCOBRA/TRAC Acceptance for AP600 Long-Term Cooling Analysis

The WCOBRA/TRAC code was used to analyze the long-term cooling portion of the AP600 plant transient. The WCOBRA/TRAC calculations characterize the long-term cooling behavior of the plant. WCOBRA/TRAC has been validated against OSU low-pressure integral systems tests that simulate the long-term cooling phenomena anticipated for the AP600 in WCAP-14776 (Reference 10).

The key parameters that are of interest include:

- Transient mass distribution in the primary system when the system is in the long-term cooling phase
- Reactor vessel inventory and behavior of the fourth-stage ADS vent valves

- The mass and energy flow of the primary system, since the flowrate and the amount of subcooling or boiling in the core affects the potential for boron plate-out on the fuel rods

- Coupled behavior between the injection source flow rate and the amount of vaporization generated in the core

- The effect of different break locations and single failure assumptions

The OSU test facility was specifically designed to model the long-term cooling portion of the AP600 transient. Sufficient instrumentation was provided to identify and quantify the long-term cooling phenomena, so that validation of WCOBRA/TRAC was accomplished. The methodology used in AP600 long-term cooling analysis cases is described in WCAP-14601 (Reference 3) and was approved by the staff for AP600.

Several issues were identified and resolved during the Staff review of the AP600 long-term cooling methodology. Foremost was the test basis for characterization of long-term cooling phenomena and the performance of WCOBRA/TRAC in predicting the tests. The scaling rationale of the OSU APEX facility during the long-term cooling phase was shown to be adequate, as were the WCOBRA/TRAC simulations of selected tests as documented in WCAP-14776. A second issue was the use of "window" mode calculations of segments of the long-term cooling transient. Using this technique the plant boundary conditions at a given time in the transient are specified as input to WCOBRA/TRAC, and the system behavior is calculated by the code for the quasi-steady-state situation under those boundary conditions. In this way, the limiting time intervals during long-term cooling can be analyzed without the need to invest in the long computer running time necessary to execute a problem for the entire long-term cooling phase. The OSU test simulations and the AP600 plant predictions were performed as windows. The Staff concluded that the WCOBRA/TRAC window mode methodology was acceptable for demonstrating the long-term cooling capability of the AP600.

2.3.3 WCOBRA/TRAC Acceptability for AP1000 Long-Term Cooling Analysis

The PIRT prepared for AP600 long-term cooling (LTC) behaviors continues to apply to the AP1000 design with no major changes, as previously noted. Reference 4 justifies that the scaling rationale of the OSU long-term cooling test facility also applies to the AP1000 plant design. Therefore, there are no additional phenomena that would require the addition of novel features to, and/or further validation of WCOBRA/TRAC for performing AP1000 long-term cooling 10CFR50.46 LOCA analyses. The simulations in WCAP-14776 predicting the OSU tests validate and justify the ability of WCOBRA/TRAC to predict the AP1000 LTC system phenomena.

The original AP600/AP1000 WCOBRA/TRAC LTC model was based on a simplified noding. In particular, the core region was subdivided in [ ]^{a,c}. Questions were raised about the adequacy of such modeling, and in particular, the axial core noding was judged to be insufficient to correctly model the core axial void fraction distribution.

As a result, the AP1000 LTC model was extended/modified as follows:

- The core was subdivided in [ ]^{a,c}.
- The core region was subdivided axially in [ ]^{a,c} and is now consistent with nodalizations used to validate WCOBRA/TRAC against G1, G2, and FLECHT-SEASET tests.
- The upper plenum explicitly models the CCFL region above the upper core plate, and the nodalization is now equivalent to the Westinghouse WCOBRA/TRAC LBLOCA model, which was validated against full-scale Upper Plenum Test Facility (UPTF) tests.

Additional code validation was identified for the application of the revised WCOBRA/TRAC model to simulate the AP1000 LTC conditions. Selected G1 and G2 full-scale boiloff tests at pressure and power levels, which are prototypical of AP1000 conditions, were selected to validate the WCOBRA/TRAC core model. This validation included the determination, via sensitivity studies, of a corrective multiplier applied to the interfacial drag model such that the average core void fraction could be accurately predicted. Results from this validation are discussed below.

The validated model was then applied to perform the LTC transient analyses for the AP1000 Design Control Document (DCD), including a DEDVI break, which exhibits the most limiting relationship between core decay power (maximum) and available PXS liquid head (minimum).

The revised WCOBRA/TRAC analysis showed that adequate core cooling exists during the entire LTC transient. The core inlet flow is more than sufficient to remove the decay heat, and additional liquid is stored in the upper plenum and hot leg. No core temperature excursion is predicted to occur.

In addition, a sensitivity study was performed where the interfacial drag coefficient was reduced by 20 percent. Results indicate that, under the AP1000 conditions, the core interfacial drag model has a negligible effect on the inner vessel mixture level. In both calculations (YDRAG=1.0 and YDRAG=0.8), mixture level is predicted in proximity of the hot leg centerline and the hot leg collapsed liquid level is almost identical in the two sensitivity cases.

These results are consistent with conclusions about the AP1000 system discussed in Reference 13. The analysis in Reference 13, based on the simple AP1000 model, showed that the system draws more flow through the core than is needed to remove decay heat. Under those circumstances, the mixture level is above the top of core and is virtually independent of the level swell model used within the core. In the AP1000 DEDVI event, during the LTC, the average core exit quality is indicated to be always less than 50 percent. This flow regime is quite different than a boiloff scenario such as in the G1 and G2 tests. In the boiloff mode, the exit quality is approximately 1.0 and, once the two-phase mixture level drops below the top of the heated section, the rods are exposed to pure steam and can undergo an almost adiabatic heatup. As a result, because of the sufficient liquid supply to the core, core heatup does not occur during the AP1000 LTC phase following a LOCA event.

WCOBRA/TRAC Core Void Fraction Model Assessment Against G1 and G2 Low-Pressure Boiloff Tests

G1 (Reference 14) test runs 28, 35, 38, 58, and 61; and G2 (Reference 15) test runs 728, 729, 730, 732, 733, and 734 were selected to validate the WCOBRA/TRAC core void fraction model used to perform the AP1000 LTC analysis. The following table shows the comparison between the test conditions and conditions expected in the AP1000 during the transient.

Test	Pressure (psia)		Power (kW/ft)		Core/Assembly Flow (in/s)		Inlet Subcooling (°F)	
AP1000	20	45	0.02	0.18	0.4	0.8	14	80
G1	[							
G2	[							

a,c

As discussed in the previous summary, the AP1000 core is not expected to be in a boiloff mode. Nevertheless, these experiments are useful to characterize the void fraction distribution and/or average void fraction within the core region when the mixture level is located above the top of the core.

G1 represents a prototypical [ ]a,b,c. G2 represents a [ ]a,b,c.

For G1, the WCOBRA/TRAC model includes the heated section, the lower plenum and the upper plenum, and the downcomer region. The heated section is subdivided in [ ]a,b,c. The boiloff test is initiated by setting the liquid level in the heated section and in the downcomer region to a given value. The power is turned on at the beginning of the test. The liquid in the lower plenum [ ]a,b,c

The WCOBRA/TRAC model for G2 is similar. In this case, [ ]a,b,c.

At each given time, the location of the mixture level is defined by examining the rod temperature axial distribution. The rod surface temperature is close to saturation below the mixture level and suddenly increases significantly above the saturation temperature above the mixture level.

The average void fraction below the mixture level is related to a parameter called swell "S" defined as follows:

a,b,c

[ ]

Figure 2-6 shows the measured swell compared to the swell predicted by the nominal WCOBRA/TRAC interfacial drag model. The swell (or average void fraction) tends to be overpredicted by the code.

The G1 and G2 calculations were repeated by applying a multiplier (YDRAG = 0.8) to the interfacial drag coefficient. Figure 2-7 shows the effect of a reduced interfacial drag. The predicted swell or void fraction is now in good agreement with the test data captured within ±20 percent. This multiplier was selected to be used in the WCOBRA/TRAC LTC analysis for AP1000.

Results from Revised WCOBRA/TRAC Model for AP1000 Long-Term Cooling Phase Following a DEDVI Break in PXS Room B

The transient begins from the end of DEDVI analysis of NOTRUMP at 3000 seconds, and continues with boundary conditions provided by WGOTHIC (containment analysis) predictions.

The results from the WCOBRA/TRAC LTC calculation are presented here. The AP1000 DCD includes a more detailed description of the transient. Here the discussion is limited to address the level swell issue and to derive some conclusions about the vessel liquid inventory, which demonstrates that adequate cooling exists during the LTC.

The time scale of the plots is adjusted to reflect DEDVI break transient time. Figure 2-8 shows the upper plenum pressure. The pressure decreases from its initial value to reach a quasi-steady-state value of 28 psia at about 7000 seconds.

Figures 2-9 and 2-10 show the ADS-4 integrated flows and the integrated flows from the DVI nozzles.

The inner vessel collapsed liquid level as well as the core region only collapsed liquid level are shown in Figures 2-11 and 2-12.

Figure 2-13 shows that the mixture level is located in proximity of the hot leg centerline.

The LTC case was analyzed with both nominal interfacial drag model and with 20-percent reduced interfacial drag model, and it was observed that the hot leg levels from these calculations were nearly identical as shown in Figure 2-14.

This result is an indication that once the mixture level is located above the top of the core and well into the upper plenum, the interfacial drag model or core swell model has a small effect on the overall system behavior.

The liquid supply (core inlet liquid flow) is always sufficient to remove the decay heat. Additional liquid is stored in the upper plenum and discharged by the ADS-4. Figure 2-15 shows that the ADS-4 average exit quality is around 50 percent during the LTC transient.

The predicted void fraction at the top of the core hot assembly is approximately 0.8 during the transient (Figure 2-16), which is another indication that sufficient liquid is provided at the top of the core preventing core heatup from occurring.

Figure 2-17 shows that the cladding temperature in the top region of the core is always close to the saturation temperature, and no heatup excursion is predicted to occur.

Additional LTC Considerations

Further investigations were made to establish what flow regime should be expected in the top region of the core to further support that under the conditions expected during the LTC, adequate core cooling is provided to prevent core heatup from occurring.

The expected flow regime at the top of the core is a churn or pulsated annular flow. The steam velocity is so low that entrainment of droplets is not expected to occur. Based on Ishii and Grolmes (Reference 16) inception criteria for droplet entrainment in two-phase cocurrent film (roll wave and liquid jet instabilities), the critical superficial velocity for droplet entrainment was estimated to be 77 ft/s (P = 40 psia). Yonomoto, et al. (Reference 17) (JAERI) established a criterion for entrainment onset based on reflood tests in rod bundle prototypical geometries and conditions. Based on the Yonomoto model, the onset is at about 20 ft/s at the same conditions. During the LTC, the vapor superficial velocity at the core exit is expected to be lower than 16 ft/s.

The possibility that the CHF could be exceeded, below the two-phase mixture level, was also investigated. Schoesse, et al. (Reference 18) presented a review of CHF correlations applicable to low upward flows near atmospheric pressure. It was found that the AP1000 typical heat flux (the average heat flux is about 1.0 Btu/ft^2-s at 3000 sec.) is less than the critical heat flux, which can be predicted with their model.

The WCOBRA/TRAC computer code is applied to the AP1000 design certification long-term cooling analysis as follows:

Code Version

A special version of the WCOBRA/TRAC computer code was created for the AP600 SSAR analysis by incorporating additional capability to model the unique features of the AP600, as documented in WCAP-14776, Section 4. A similar approach is used to perform the AP1000 design certification long-term cooling LOCA analysis. The same updates identified in WCAP-14776, Section 4 as being added to WCOBRA/TRAC for the AP600 analysis are included in the creation of an "AP" version to perform the AP1000 long-term cooling licensing analysis. The "AP" version of WCOBRA/TRAC includes the discretionary and non-discretionary code changes that have been made since the AP600 SSAR analysis was performed, which constitute the "2000 formulation" of the code and which have been reported to the NRC by Westinghouse (Reference 11) per the 10CFR50.46 annual reporting process. The details of the

code changes made since the AP600 analysis was performed are provided in Appendix A. The impact of implementing any or all of the changes in WCOBRA/TRAC is judged to be minor on the simulations of AP600 long-term cooling scenarios because they deal primarily with large break LOCA-related phenomena. The "AP" code version also includes logic to enable the user to specify a multiplier (YDRAG) to the COBRA channel interfacial drag coefficient computed by the code.

Core Nodalization Scheme

In WCOBRA/TRAC simulations, the degree of detail used in the axial noding of the active fuel region influences the calculated core collapsed liquid levels. To establish an appropriate noding scheme, tests from the G1 and G2 test facilities were simulated to validate WCOBRA/TRAC for the prediction of the core mixture level swell and other pertinent phenomena over the AP1000 long-term cooling range of conditions, as previously discussed in this subsection.

The WCOBRA/TRAC core nodalization validation against level swell test data justifies that the same core axial noding detail used in the AP1000 large break LOCA analysis may be applied in long-term cooling predictions. The long-term cooling analysis noding scheme is shown as Figure 2-18. The same number of cells is specified axially in the active fuel region section channels as in Reference 5. The core radial nodalization from the AP1000 large break LOCA WCOBRA/TRAC model is used to model radial power effects in the long-term cooling simulations. Channels 10, 23, 26, and 39 represent [

]^{a,c}.

Another outcome of the G1/G2 WCOBRA/TRAC validation is the identification of a core interfacial drag multiplier (YDRAG) value of 0.8 as the means to obtain a good prediction of the level swell test results. In the AP1000 DCD analyses, the YDRAG parameter is set to 0.8 in all core channels at all elevations.

Upper Plenum Nodalization Scheme

To enhance the upper plenum flow pattern prediction capability, the noding scheme in Section 3 of the WCOBRA/TRAC AP1000 LTC model in Figure 2-18 has adopted the large break LOCA nodalization. In this way, the upper plenum noding corresponds to that used to successfully simulate the UPTF facility upper plenum tests in Reference 2 to provide the justification for the large break LOCA upper plenum modeling. The model includes [

]^{a,c} as depicted in Figure 2-18. Channels 50, 51, 52, and 53 represent [

]^{a,c}, respectively.

In Section 4, at the hot leg elevation, all of the Section 3 channels [

]^{a,c} The upper plenum channels specified for the region between the top of the active fuel and the hot leg bottom elevation (Section 3) contain [

]^{a,c}.

AP600 FSER Restrictions

The AP600 FSER (Reference 8) identified three restrictions on the approval of WCOBRA/TRAC for AP600 long-term cooling analyses. The FSER-specified restrictions follow, together with the means by which the AP1000 analysis will comply with each:

1. Westinghouse shall ensure the nodalization of the AP600 design long-term cooling model corresponds to that used in the OSU calculations: the WCOBRA/TRAC nodalization for AP1000 LTC analyses was extended/modified as described above to be consistent with the nodalization used to validate WCOBRA/TRAC against test data.

2. Westinghouse shall ensure the window time span results in a quasi-steady -state solution: the window mode long-term cooling computations performed for AP1000 will be executed until the quasi-steady state condition is achieved. The AP600 SSAR Subsection 15.6.5.4C long-term cooling analyses were performed using "windows" to investigate ECCS performance at the most limiting time intervals during post-LOCA core cooling. The window mode analysis technique is also used in the AP1000 long-term cooling design certification cases. In addition, the case in which sump recirculation occurs earliest in time among the AP1000 long-term cooling transients (a double-ended DVI (DEDVI) LOCA break which drains the IRWST directly to containment) is analyzed from the start of long-term cooling until containment recirculation is established. This continuous calculation technique is compared with the window mode approach result for the DEDVI break for the time interval bracketing the start of recirculation in the preliminary analysis presented in Section 3.3.3 of the AP1000 Plant Description and Analysis Report. The comparison of results shows that the WCOBRA/TRAC predictions are equivalent whether the code is run continuously or with the window-mode approach.

3. Westinghouse shall ensure the code is not applied outside "the corresponding parameter range from the OSU experiments. In particular, WCOBRA/TRAC is not validated for core dryout and heatup." The design of the AP1000 Passive Core Cooling Systems prevents core uncovery and heatup from occurring during long-term cooling phase of design basis accidents.

Major Issues

The NRC staff approval of the WCOBRA/TRAC long-term cooling calculational methodology was specific to the AP600 design, based on the parameter range of the OSU experimental validation. WCOBRA/TRAC was not considered valid for the prediction of core dryout and heatup phenomena during long-term cooling because the OSU tests simulated by Westinghouse did not exhibit any core heatup or dryout phenomena. Further, the noding detail of the core and upper plenum regions was called into question during the AP1000 licensing review.

Resolution of Issues

No core uncovery and no fuel rod heatup are predicted to occur in the AP1000 DCD analysis of the limiting case condition. Based on this confirmation of the AP1000 system design, no core uncovery or fuel rod heatup is anticipated to occur in the long-term cooling phase of any postulated scenario in the

AP1000 design basis scope. The LTC analyses are conservative because they are performed in accordance with Appendix K of 10CFR50.

The WCOBRA/TRAC nodalization used in the AP1000 design certification long-term cooling ECCS performance analyses is expanded in response to the NRC review questions, consistent with level swell model validation against G1 and G2 data.

Conclusions

The PIRT and scaling review of WCAP-15613 shows that there are no new phenomena involved in the AP1000 safety performance during LTC relative to AP600, and that the OSU facility data continue to apply to the AP1000. Therefore, the validation of the WCOBRA/TRAC computer code for post-LOCA long-term cooling analysis against the OSU data applies to AP1000 as well. Additional comparison of WCOBRA/TRAC against test data has validated the core void fraction model. The WCOBRA/TRAC nodalization for AP1000 has been extended/modified to be consistent with the WCOBRA/TRAC validation bases. The WCOBRA/TRAC computer code and nodalization scheme described herein is used to perform the AP1000 design certification long-term cooling ECCS performance analysis.

2.3.4 Post-Accident Core Boron Concentration

An evaluation has been performed of the potential for the boron concentration to build up in the core following a cold leg LOCA. The evaluation methodology, simplified calculations, and their results are discussed in Appendix H. This evaluation considers both short-term operations, before ADS is actuated, and long-term operations, after ADS is actuated. These evaluations and their results are discussed in the follow paragraphs.

Short-Term – Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the RCS and prevents buildup in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and PRHR HX has to stop. In addition, significant injection of borated water is needed from the CMTs and the CVS. For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void, the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

Short-Term Results – As shown by NOTRUMP analysis, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA. The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes more than 3 hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level.

Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are guided to take an RCS boron sample within 3 hours of the accident and several more during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin and that the RCS boron concentration has not built up to excessive levels. The maximum calculated core boron concentration 3 hours after a LOCA without ADS actuation is less than 16,000 ppm. Operator action within 3 hours maintains the maximum core boron concentration well below the boron solubility limit for the core inlet temperatures during the cooldown.

Long-Term – Once ADS is actuated, water carryover out the ADS-4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The design of the AP1000 facilitates water discharge from the hot legs as follows:

- PXS recirculation flow capability tends to fill the hot legs and bring the water level up to the ADS-4 inlet.
- ADS-4 lines discharge at an elevation 3 to 4 feet above the containment water level.

With water carried out ADS-4, the core boron concentration increases until the boron added to the core in the safety injection flow equals the boron removed in the water leaving the RCS through the ADS-4 flow. The lower the ADS-4 vent quality, the lower the core boron concentration buildup.

Long-Term Results – Analyses have been performed (Appendix G) to bound the maximum core boron concentration buildup. These analyses demonstrate that highest ADS-4 vent qualities result from the following:

- Highest decay heat levels
- Lowest PXS injection/ADS-4 vent flows, including high line resistances and low containment water levels

The WCOBRA/TRAC long-term cooling analysis is consistent with these assumptions. The ADS-4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With the maximum ADS-4 vent qualities, the maximum core boron concentration peaks at a value of about 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS-4 vent quality decreases, reaching 5000 ppm about 9 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F (at 7400 ppm) and quickly drops to 40°F (at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is normally expected to be above these solubility temperatures. Even considering the minimum IRWST temperature permitted by the Technical Specifications (50°F), the minimum core inlet temperature is greater than the solubility temperature considering heatup of the injection by steam condensation in the downcomer and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration of water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

With high decay heat values, the ADS-4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular for more than 30 days. The annular flow regime moves water up and out the ADS-4 lines. This flow regime is based on the Taitlel-Dukler vertical flow regime map. Lower decay heat levels can be postulated later in time or just after a refueling outage. Significantly lower decay heat levels result in lower ADS-4 vent qualities. They also result in ADS-4 vent flows/velocities that are lower. Even with low ADS-4 vent flow velocities, the AP1000 plant will move water out the ADS-4 operating as a manometer. Small amounts of steam generated in the core reduce the density of the steam/water mixture in the core, upper plenum, and ADS-4 line as it bubbles up through the water. As a result, the injection head is sufficient to push the less dense, bubbly steam/water mix out the ADS-4 line.

At the time recirculation begins, the containment level will be about 109.3 feet (for a non-DVI LOCA) and will be about 108.0 feet (for a DVI LOCA). Over a period of weeks after a LOCA, water may slowly leak from the flooded areas in containment to other areas inside containment that did not initially flood. As a result, the minimum containment water could decrease to 103.5 feet. During recirculation operation following a LOCA and ADS actuation, the operators are guided to maintain the containment water level above the 107-foot elevation by adding borated water to the containment. In addition, if the plant continues to operate in the recirculation mode, the operators are guided to increase the level to 109 feet within 30 days of the accident. These actions provide additional margin in water flow through the ADS-4 line. The operators are also guided to sample the hot leg boron concentration prior to initiating recovery actions that might introduce low temperature water to the reactor.

2.4 ASSESSMENT OF DG-1096 RELATED ISSUES

In a workshop (April 9, 2001) held to discuss Draft Regulatory Guide DG-1096, several attributes were discussed which should be considered in determining the extent to which the DG process should be used in the development, assessment, and application to an evaluation model. These are:

- Novelty of the evaluation model compared to the currently acceptable model.
- The complexity of the event being analyzed.
- The degree of conservatism of the evaluation model.
- Risk or safety importance of the event.

For the AP1000 analysis program, the WCOBRA/TRAC-AP code contains the models and correlations which were reviewed by the NRC staff and approved as comprising a best-estimate large break LOCA code for 3-loop and 4-loop Westinghouse plants consistent with the guidance provided in Regulatory Guide 1-157. WCOBRA/TRAC was later approved for the AP600 large break LOCA analysis application in Reference 8. There are no novel changes in WCOBRA/TRAC-AP for large break LOCA analysis

relative to the code version approved for AP600, only the discretionary and non-discretionary changes delineated in Appendix A. Therefore, WCOBRA/TRAC-AP has already undergone the type of review envisioned for a best-estimate large break LOCA analysis computer code in DG-1096.

The DG-1096 attributes are discussed for the AP1000 LTC application of WCOBRA/TRAC-AP below:

- The code version to be utilized for the AP1000 program is the same as that utilized for the AP600 program with the discretionary and non-discretionary changes being implemented as discussed in Appendix A of this document and the addition of the multiplier on the interfacial drag coefficient.

- The AP1000 long-term cooling period is not considered to be a complex event. Nevertheless, this event and WCOBRA/TRAC were thoroughly reviewed for application to the AP600 plant design. Analyses with the approved AP600 code version did not indicate the existence of new phenomena for the AP1000 design compared to those observed for the AP600 design during LTC.

- The LTC evaluation model and methodology continue to be based on the use of Appendix-K required features. As such, the result will be a conservative calculation with respect to the expected plant response.

- The AP1000 analyses indicate that margin to core uncovery exists during the limiting LTC event. As such, significant margins to the 10 CFR 50.46 limits exist for this plant design.

2.5 REFERENCES

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2. WCAP-12945-P-A, Volumes 1-5, "Code Qualification Document for Best-Estimate LOCA Analysis," Bajorek, S. M., et al., 1998.

3. WCAP-14601, Revision 2, "AP600 Accident Analyses - Evaluation Models," 1998.

4. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

5. WCAP-14171, Rev. 2, "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant-Accident," Hochreiter, L. E., et al., March 1998.

6. Siemens Report E314/90/17, "Upper Plenum Test Facility, Test 21," Seimens, A. G. (KWU), 1990.

7. JAERI-Memo 59-446, "Data Report on Large-Scale Reflood Test-78 (RUN 058)," Okubo, T., et al., 1985.

8. NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," U.S. Nuclear Regulatory Commission, September 1998.

9. AP600 Design Control Document (DCD), Westinghouse Electric Corporation, 1999.

10. WCAP-14776, Revision 4, "WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report," Garner, D. C., et al., March 1998.

11. Letter LTR-NRC-01-6 from H. A. Sepp, Westinghouse to J. S. Wermiel, USNRC, "10CFR50.46 Annual Notification and Reporting for 2000," March 13, 2001.

12. WCAP-15612, "AP1000 Plant Description and Analysis Report," December 2000.

13. AP1000 DSER Open Item 21.5-3 response.

14. WCAP-9764, "Documentation of the Westinghouse Core Uncovery Tests and Small Break Evaluation Model Core Mixture Level Model," July 1980.

15. Andreychek, T. S., "Heat Transfer above the Two-Phase Mixture Level under Core Uncovery Conditions in a 336 Rod Bundle," Volumes 1 and 2, EPRI Report NP-1692 (January 1981).

16. Ishii, M. and Grolmes, M. A. (1975), Inception Criteria for Droplet Entrainment in Two-Phase Concurrent Film Flow, A.I.Ch.E. Jl 21, 308.

17. Yonomoto, T., et al. (1987), Liquid Entrainment for Liquid Entrainment in Reflooding Phase of LOCA, J. of Nuclear Science and Technology, Vol. 24 [10].

18. Schoesse, T., et al. (1997), Critical Heat Flux in a Vertical Annulus under Low Upward Flow and near Atmospheric Pressure, J. of Nuclear Science and Technology, Vol. 34 [6].

Table 2-1 Assessment of the AP1000 LBLOCA Processes

LOCA Process	AP1000 Uniqueness WRT W Plants	WC/T Validation Does It Exist	AP1000-Specific Validation Needed	Comments
BLOWDOWN				
Critical flow	None	Yes	None	
Post-critical heat flux heat transfer Transient critical heat flux Rewetting Film boiling	None	Yes	None	
Structure heat transfer	Yes, internals	Yes, not AP1000-specific	Not needed, code can calculate	
Accumulator mixing	None	Yes	None	
Accumulator bypass	None	Yes	None	
2φ differential pressure in loops	None	Yes	None	
SG heat transfer	None	Yes	None	
High-head safety injection	Yes, CMT delivery, behavior	No	Not needed, code can calculate	Very little CMT delivery occurs before PCT is calculated
Pump 2φ behavior	Yes, canned rotor	No	No	Homologous curve data used in WCOBRA/TRAC

Table 2-1 Assessment of the AP1000 LBLOCA Processes (cont.)

LOCA Process	AP1000 Uniqueness WRT W Plants	WC/T Validation Does It Exist	AP1000-Specific Validation Needed	Comments
REFILL/REFLOOD				
ECCS bypass entrainment	Yes, accumulator delivery in downcomer	No	No; validation for AP600 applies to AP1000	Model UPTF, CCTF downcomer injection tests
Noncondensable gas effect	None	Yes	None	LOFT, W steam/water mixing
Post-CHF heat transfer	None	Yes	None	
Structural heat transfer	Yes, internals	Not specific	None	Not needed, code can calculate
Safety Injection	Yes, delivery into downcomer	No	No; validation for AP600 applies to AP1000	No CMT delivery during this period
Steam generator behavior	None	Yes	None	
Two-loop differential pressure	None	Yes	None	
REFLOOD				
Safety Injection	Yes, downcomer delivery	No	No; validation for AP600 applies to AP1000	UPTF, CCTF downcomer DVI injection data exist
Accumulator behavior	Long-term delivery	Yes, short-term for LOFT	No	LOFT data provides verification; other plant data available
Core heat transfer	None	Yes	None	
Structure heat transfer	Yes, internals	Not specific	Not needed	LOFT test had structures typical of a PWR
SG effects	None	Yes	None	
Vessel/de-entrainment	None	Yes	None	
Pump differential pressure	Yes, canned rotor	Yes, other pumps	None	Pump is a known resistance

Table 2-2 WCOBRA/TRAC Validation
Core Heat Transfer
FLECHT-SEASET reflood
FLECHT COSINE reflood
FLECHT SKEW reflood
G2 reflood
FEBA reflood
G1 blowdown
G2 blowdown
Oakridge National Laboratory film boiling
Fuel Rod Cladding Materials/Nuclear Rod
NRU reflood
NRU materials test
Fluid Mechanics Two-Phase Flow
Creare 1/15-, 1/5-scale ECCS bypass
UPTF ECCS bypass cold leg injection
1/14-, 1/3-scale cold-leg steam/water mixing
UPTF cold leg steam/water mixing
APWR 2-Phase pressure drop
Marviken critical flow tests
UPTF upper plenum de-entrainment
WCOBRA/TRAC Systems Effects Tests Verification
System Response
LOFT L2-2, L2-3, L2-5, LB-1
Semiscale mod 3 series 7
CCTF cold-leg injection tests
SCTF cold-leg injection tests

Table 2-3 Long-Term Cooling Processes

Long-Term Cooling Process	AP1000 Uniqueness WRT W Plants	Long-Term Cooling Model/ Verification Does It Exist	AP1000 Specific Validation Needed	Comments
Natural circulation	Loop 2-phase natural circulation	Yes, but not with passive safety systems	No, systems data available on AP600 specific geometry apply to AP1000	Frictional/elevation heads of the OSU APEX facility are properly scaled for AP1000
	Multiple breaks	No		
	Water delivery into downcomer	No		
	Mass distribution	No		Code can handle multiple breaks and flow paths
Long-term core heat removal	Gravity feed natural circulation, possible 2-phase flow effects	No, not in AP1000 configuration	No, OSU APEX integral systems test data apply to AP1000	Code can handle multiple breaks and flow paths; OSU systems test data has been used for code validation
Long-term SG heat removal	Yes, ADS reduces flow to SG	Yes, FLECHT-SEASET ROSA-IV, LOFT, Semiscale	None	AP1000 is less sensitive to SG behavior than conventional PWR

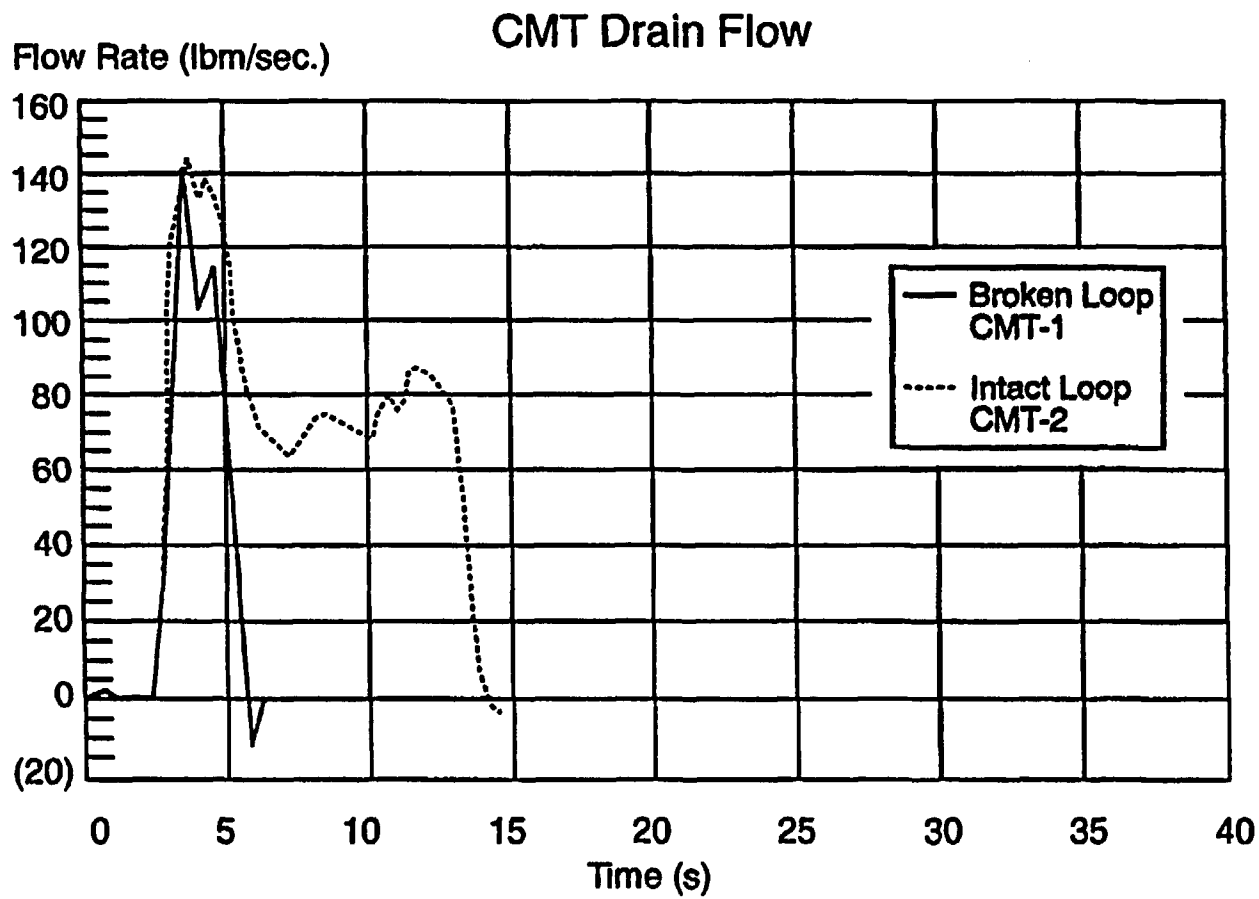


Figure 2-1 CMT Injection for an AP600 LBLOCA

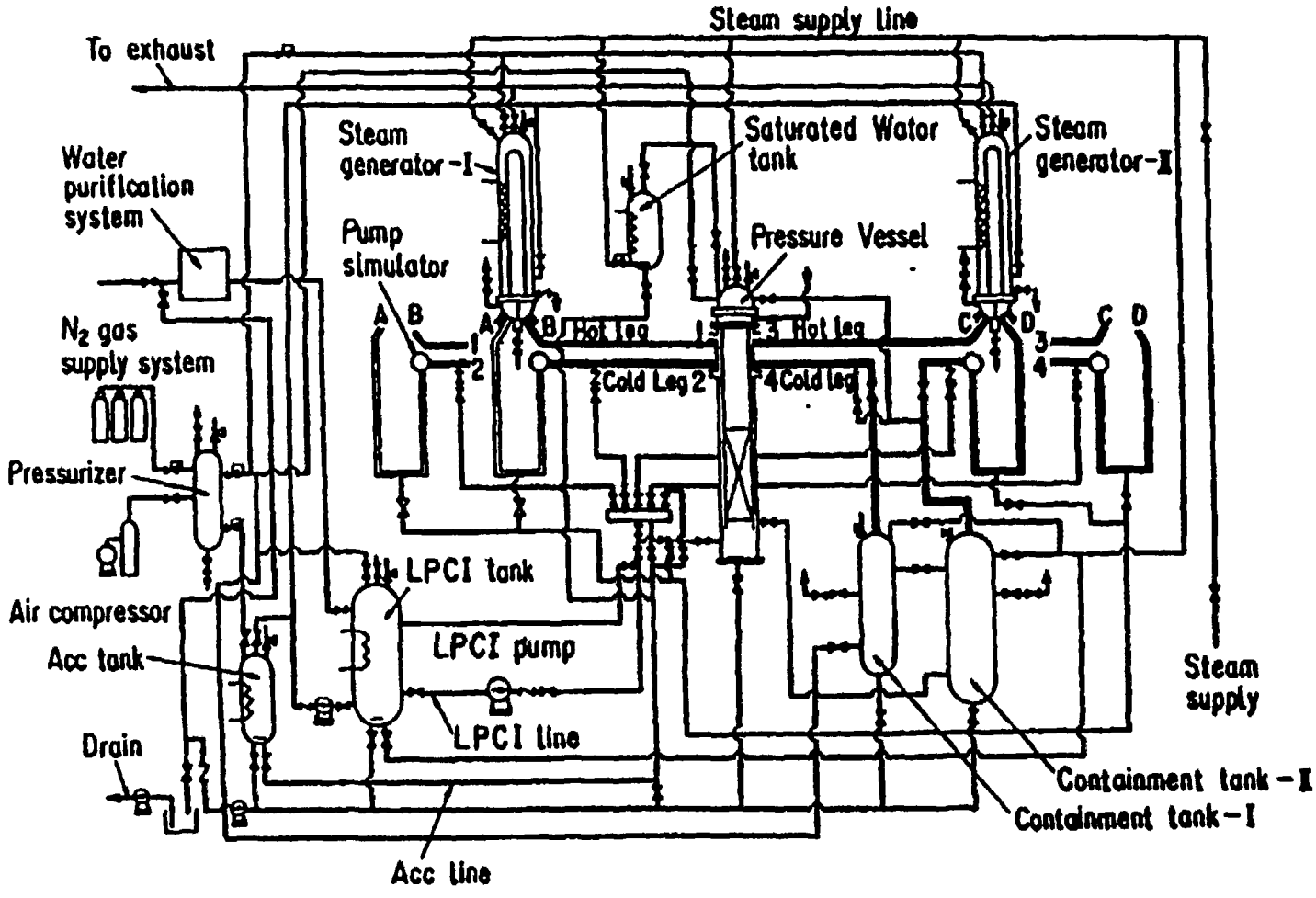


Figure 2-2 Japanese CCTF Test Facility

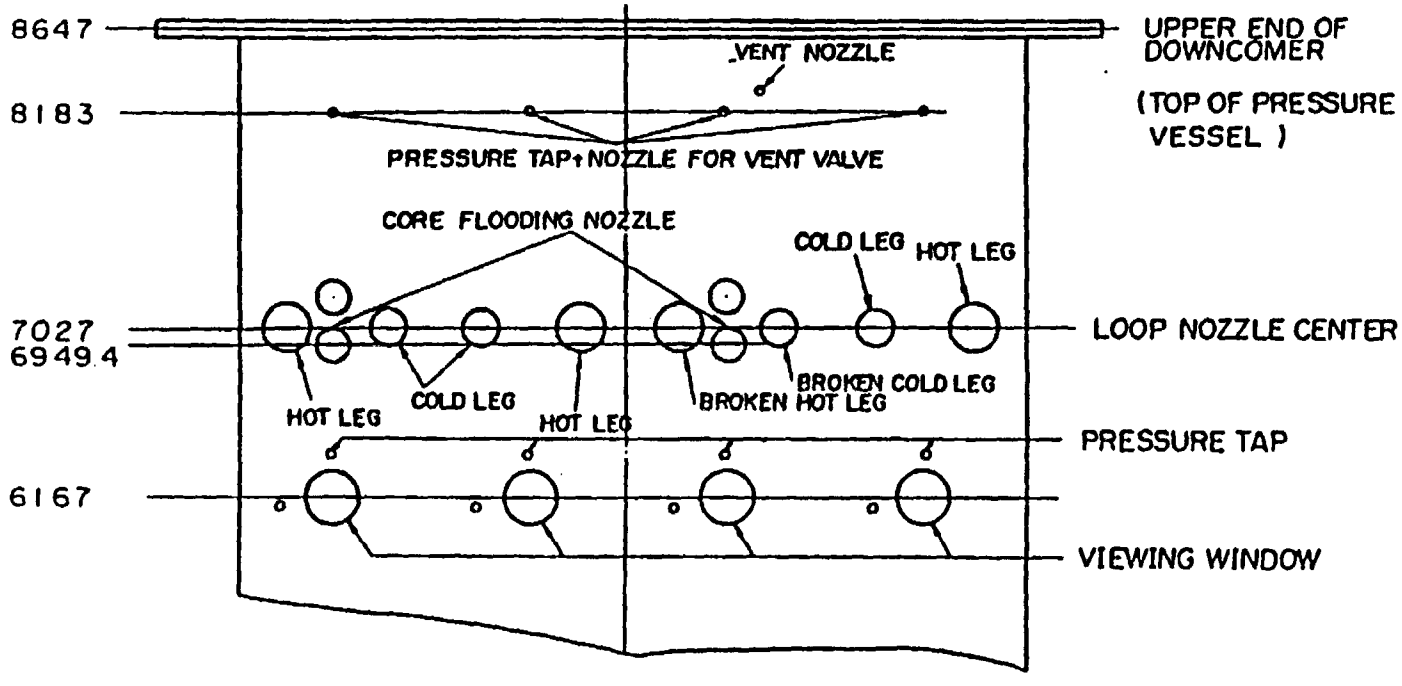
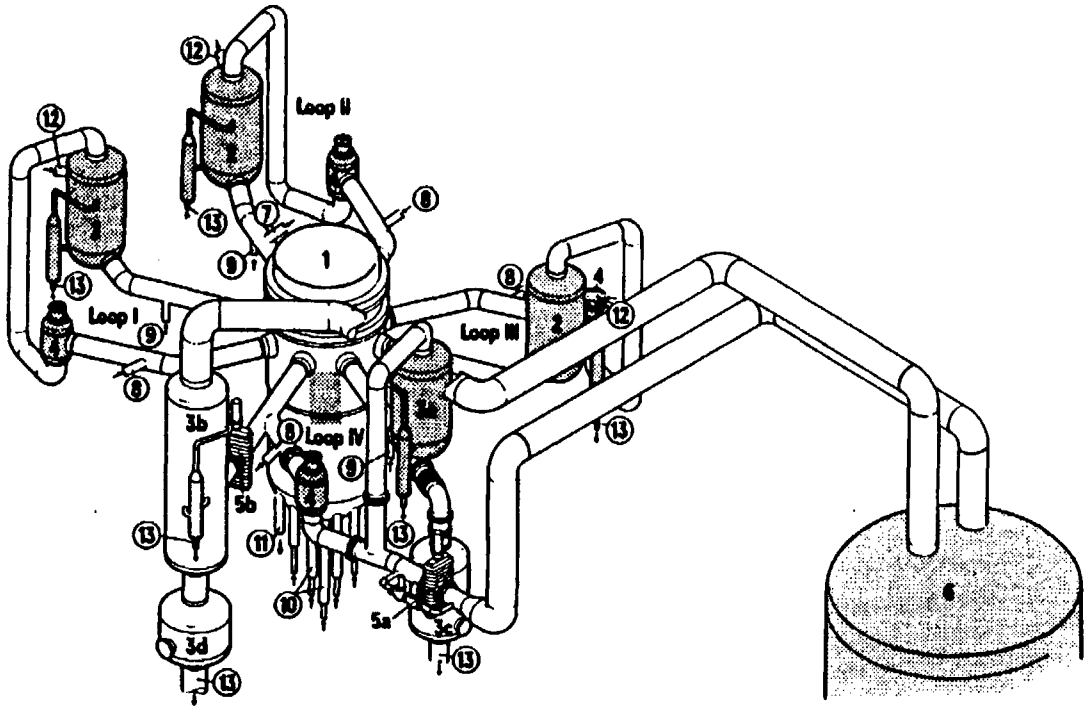


Figure 2-3 Japanese CCTF Vessel and Downcomer Geometry

SIEMENS



1 Test Vessel
2 Steam Generator Simulator (Intact Loop)
3a Steam Generator Simulator/ Water Separator (Broken Loop Hot Leg)
3b Water Separator (Broken Loop Cold Leg)
3c Drainage Vessel for Hot Leg
3d Drainage Vessel for Cold Leg
4 Pump Simulator
5a Break Valve (Hot Leg)
5b Break Valve (Cold Leg)
6 Containment Simulator

(7) Surgeline-Nozzle
(8) ECC-Injection Nozzles (Cold Leg)
(9) ECC-Injection Nozzles (Hot Leg)
(10) Core Simulator Injection Nozzle
(11) TV-Drainage Nozzle
(12) Steam Injection Nozzle
(13) Drainage Nozzle

Simulator

Figure 2-4 Upper Plenum Test Facility

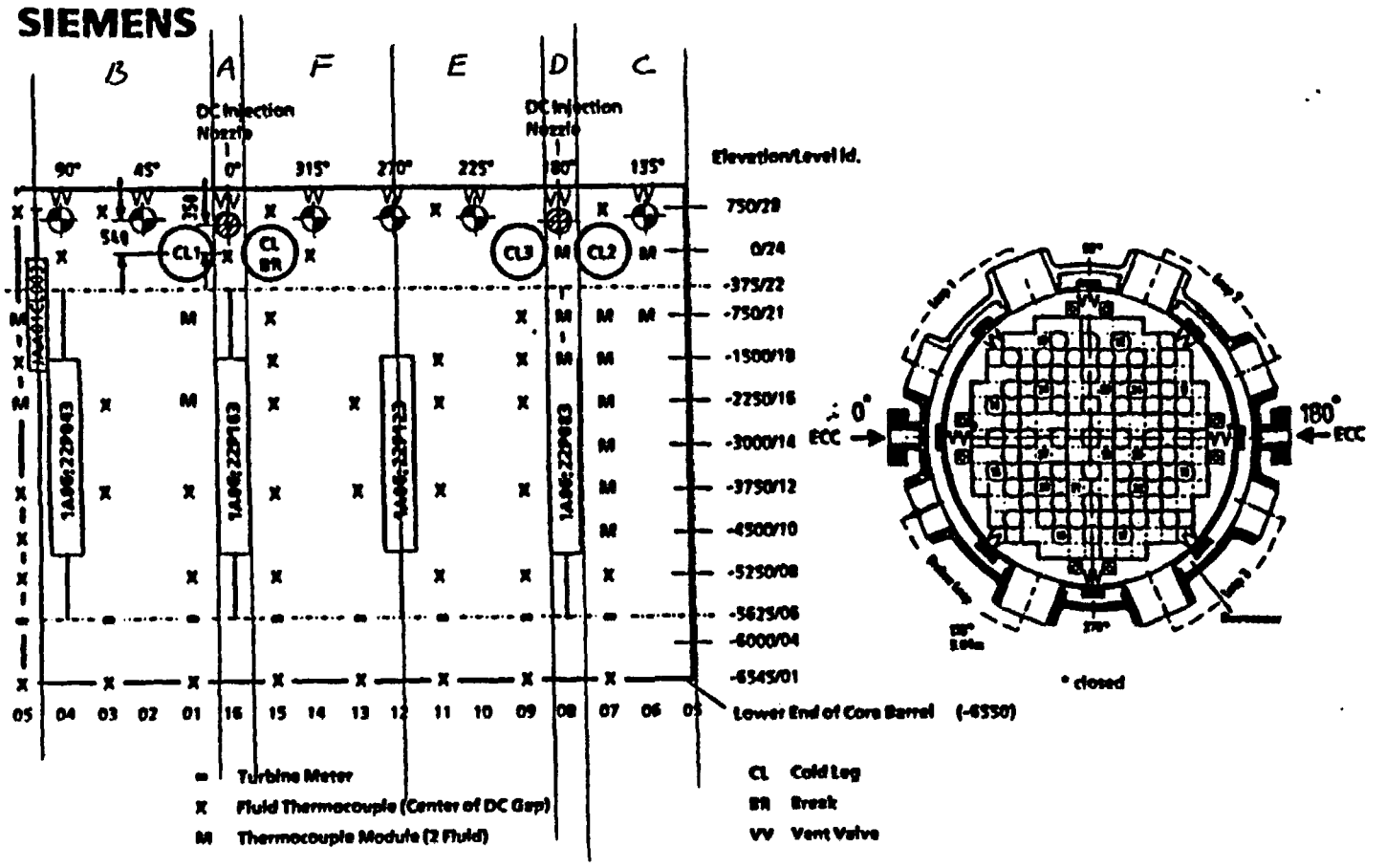


Figure 2-5 Upper Plenum Test Facility Vessel and Downcomer Geometry

a,b,c

Figure 2-6 WCOBRA/TRAC Swell Predications with Nominal Interfacial Drag

a,b,c

Figure 2-7 WCOBRA/TRAC Swell Predications with 20-Percent Reduced Interfacial Drag (YDRAG = 0.8)

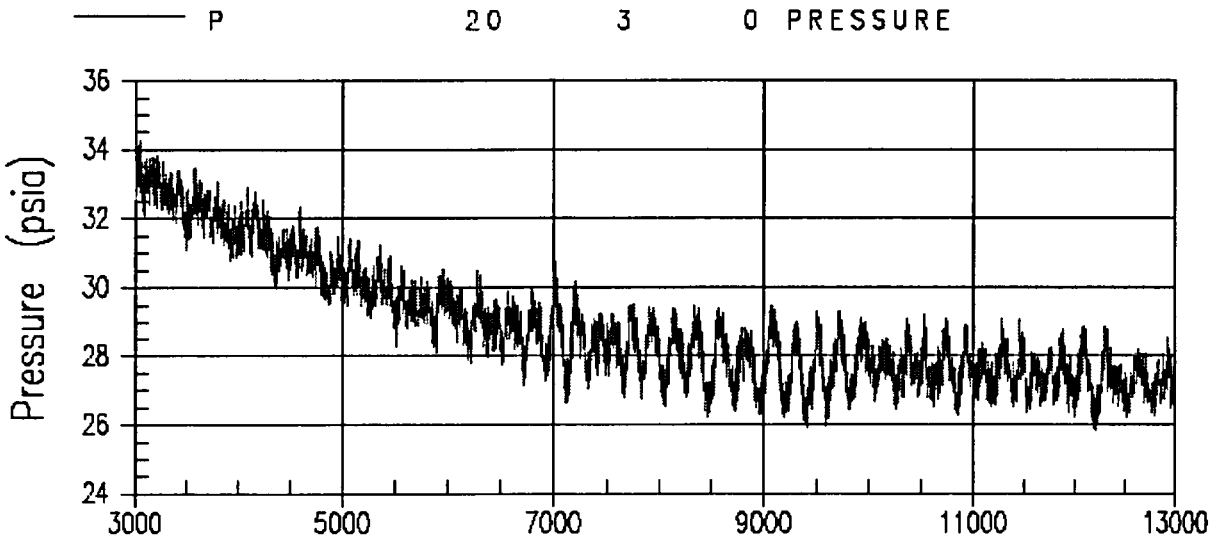


Figure 2-8 Upper Plenum Pressure

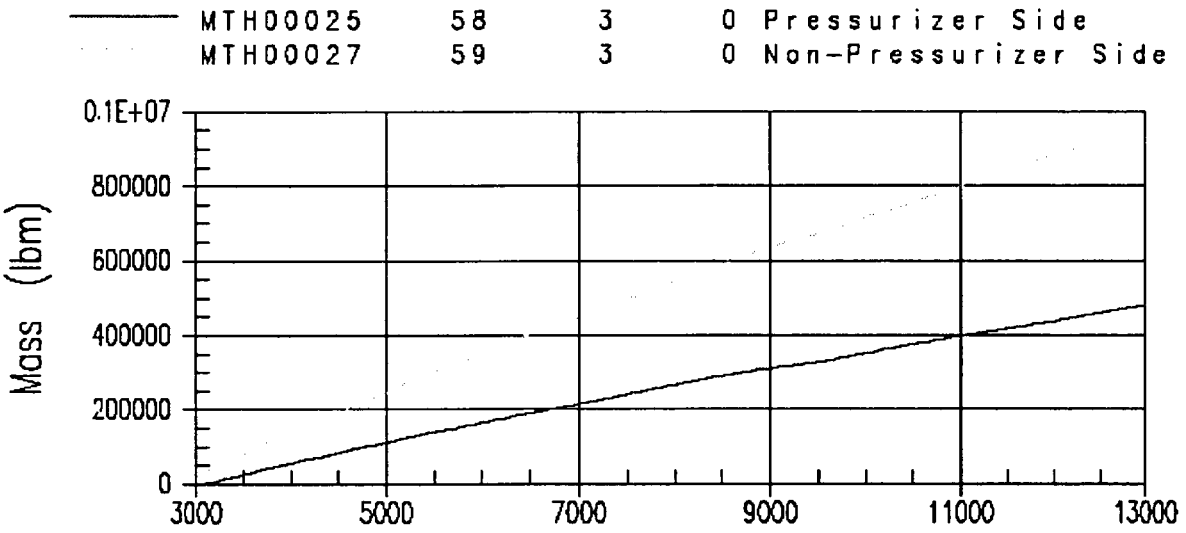


Figure 2-9 Integrated ADS4-1 and ADS4-2 Flows

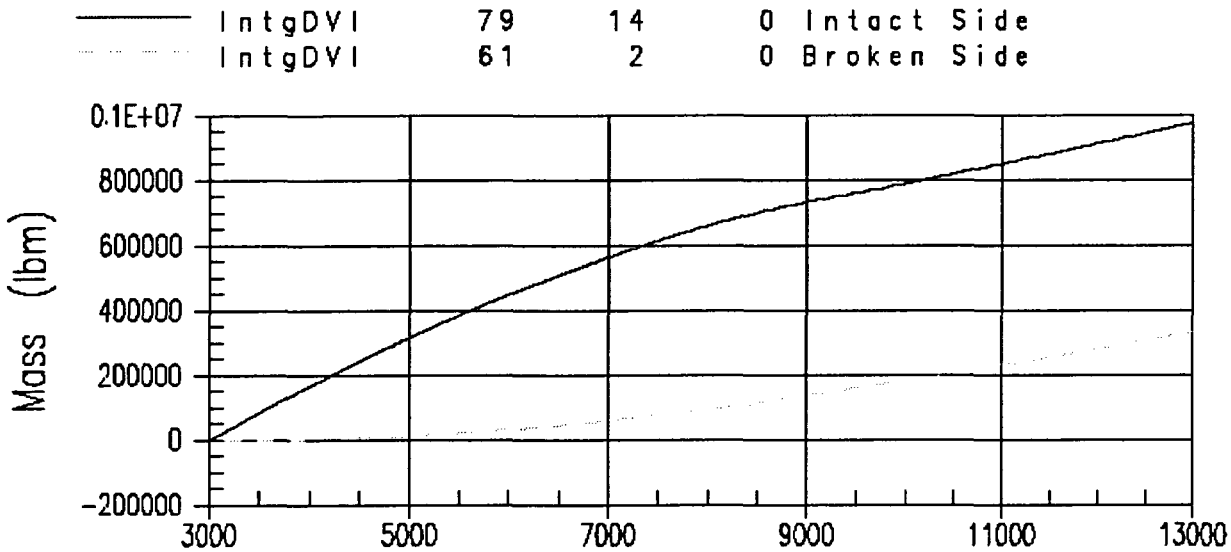


Figure 2-10 Integrated DVI Injection Flow

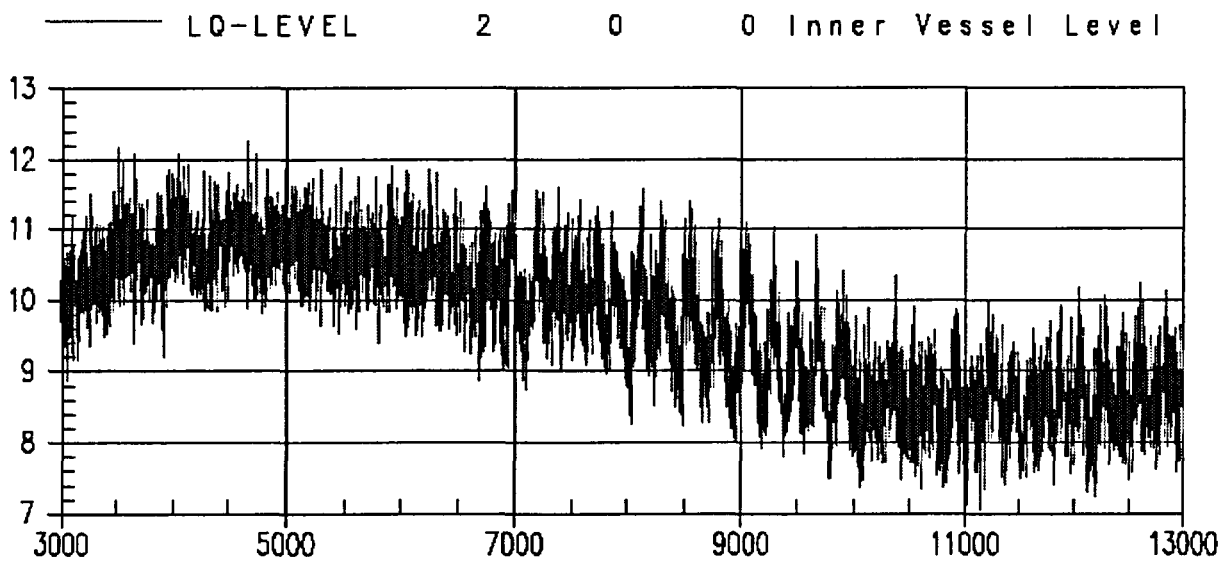


Figure 2-11 Inner Vessel Collapsed Liquid Level Above Bottom of Active Fuel

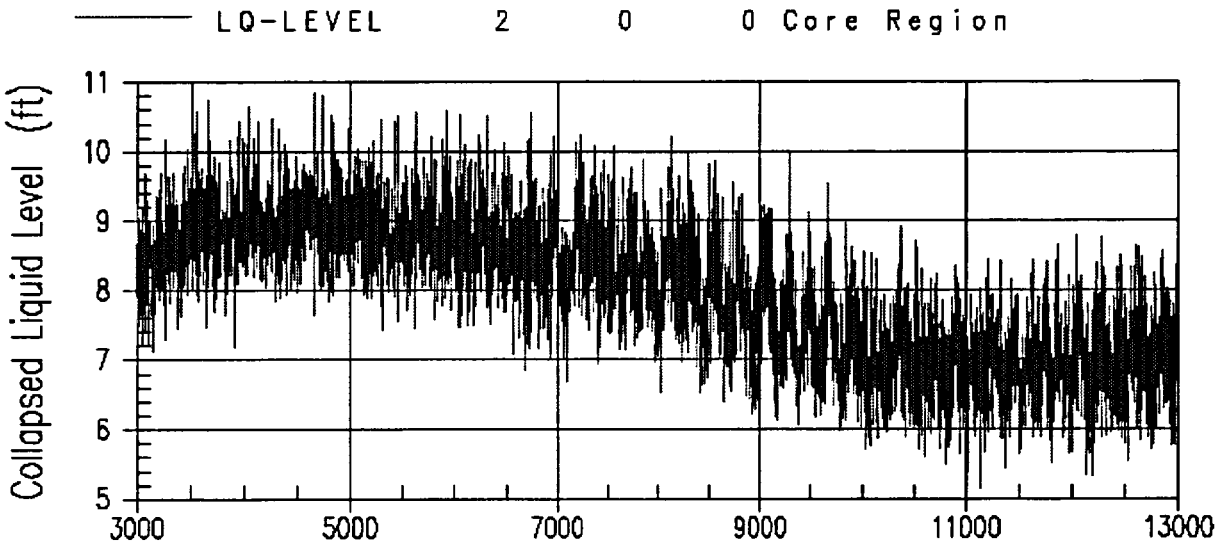


Figure 2-12 Core Collapsed Liquid Level Above Bottom of Active Fuel

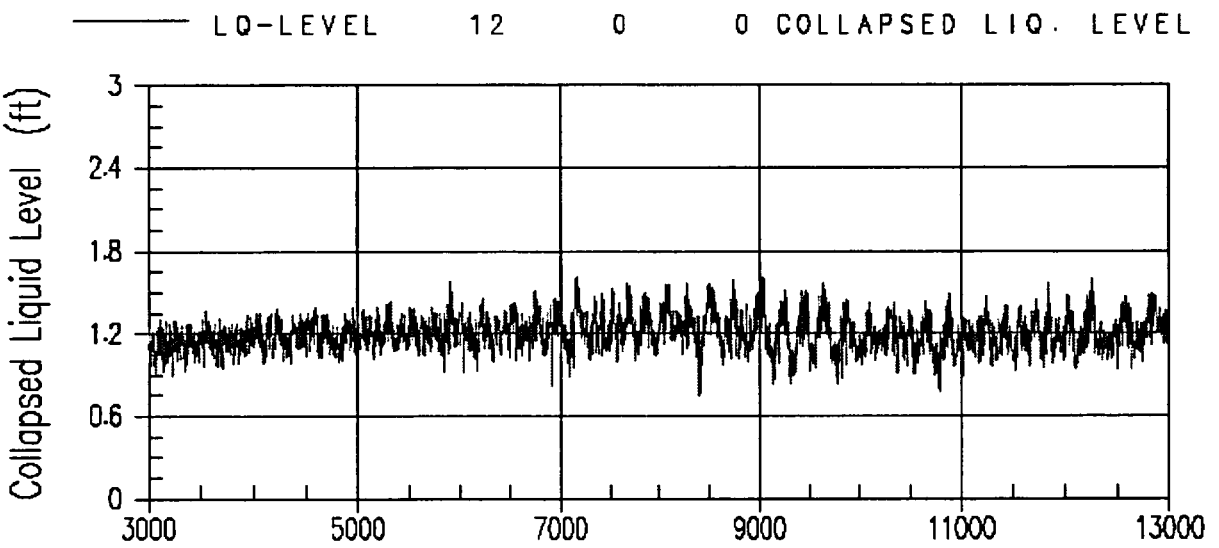


Figure 2-13 Hot Leg 2

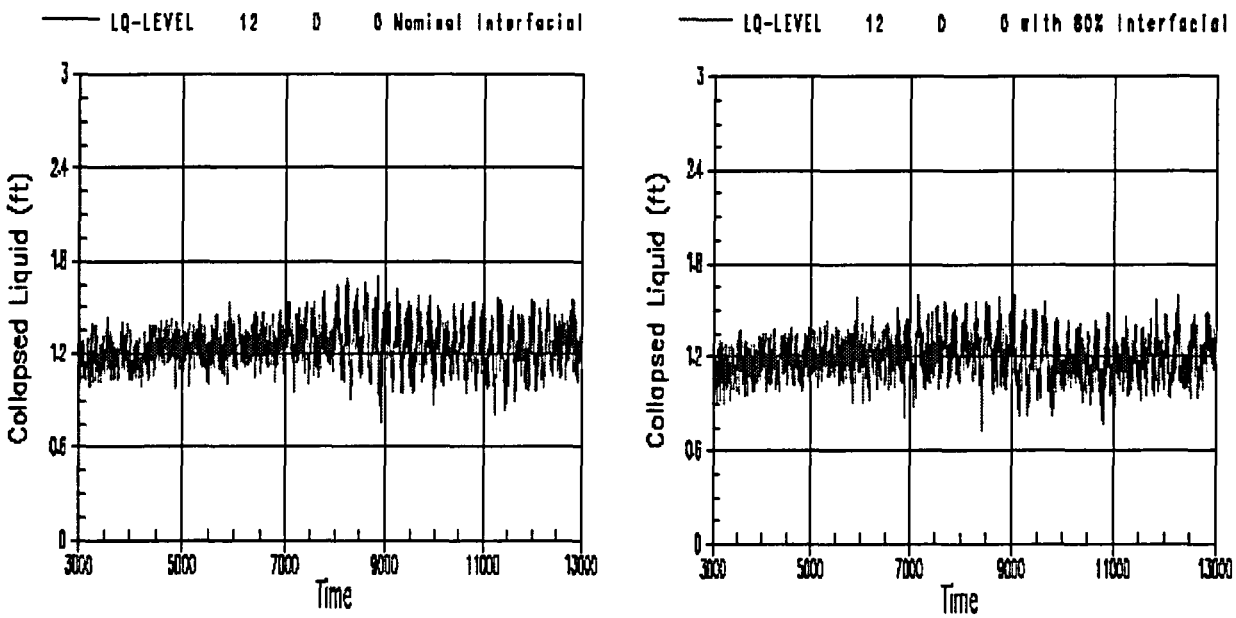


Figure 2-14 Hot Leg 2 with Nominal Interfacial Drag, and Hot Leg 2 with 80-Percent Interfacial Drag

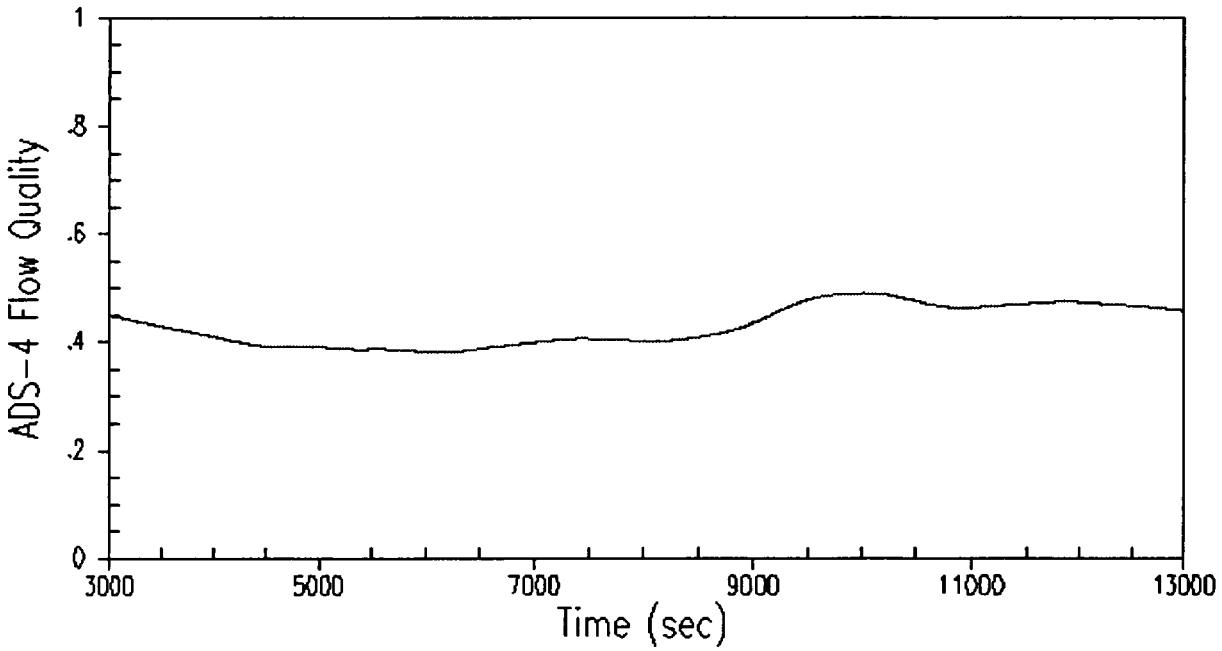


Figure 2-15 ADS-4 Exit Quality

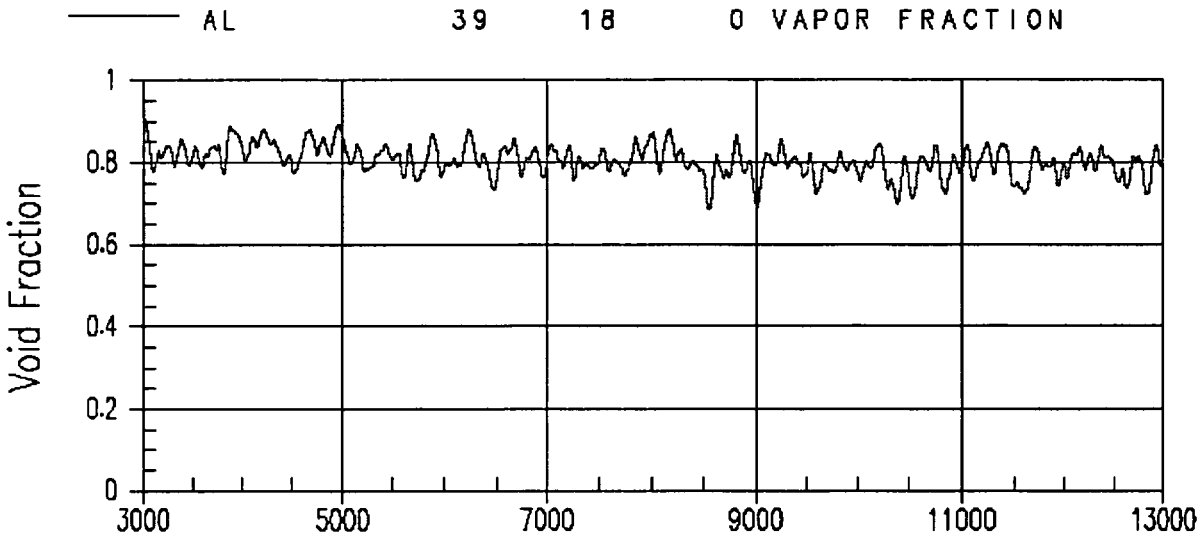


Figure 2-16 Void Fraction – TOP of Hot Assembly

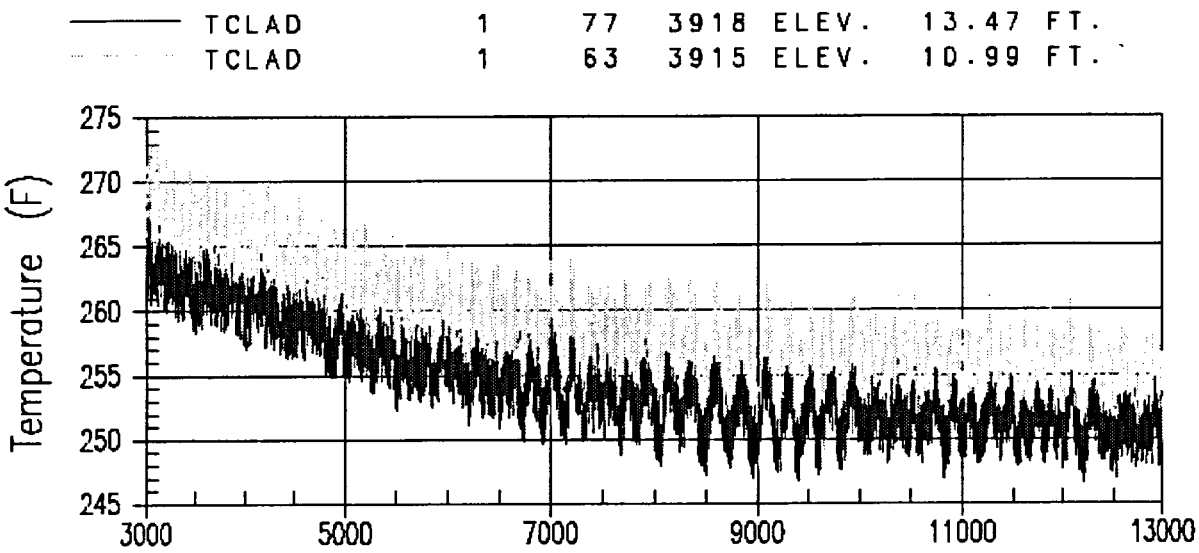


Figure 2-17 Cladding Temperatures at Higher Elevations

a,c

Figure 2-18 AP1000 Long-Term Core Cooling Vessel Noding

3.0 NOTRUMP VALIDATION FOR SMALL BREAK LOSS-OF-COOLANT ACCIDENT

3.1 BACKGROUND

The NOTRUMP code used for the AP600/AP1000 calculations consists of the modeling features that meet the requirements of Appendix K to 10CFR Part 50. The NOTRUMP code as documented in WCAP-10079-A and WCAP-10054-A (References 1 and 2), was previously approved by the NRC for small break LOCA (SBLOCA) analyses on conventional Westinghouse Pressurized Water Reactors (PWRs). The acceptance criteria for Emergency Core Cooling Systems (ECCS) for light-water nuclear power reactors, given in 10CFR50.46, require that ECCS performance be calculated in accordance with an acceptable evaluation model. Two approaches may be taken to demonstrate that an acceptable model has been applied to an ECCS design. In one approach (commonly referred to as a "best estimate"), the evaluation model must contain sufficient supporting justification to show that the analytical technique realistically describes the behavior of the reactor system during a LOCA. This necessitates comparisons to applicable experimental data along with identification and assessment of uncertainty in the analysis methods and inputs so that the uncertainty in the calculated results can be estimated. This uncertainty must then be accounted for in subsequent calculations. Alternatively, an ECCS evaluation model may be developed in conformance with the required and acceptable features of 10CFR Part 50, Appendix K, and ECCS evaluation models. Westinghouse chose to demonstrate the acceptability of the SBLOCA response of the AP600 passive reactor design using an Appendix K ECCS evaluation model.

To support this effort, a version of the NOTRUMP code, modified for the AP600 application, was developed and is documented in WCAP-14807, "NOTRUMP Final Verification and Validation Report" (Reference 3). Modifications performed to the basic NOTRUMP model enabled proper analysis of the AP600 and the supporting test matrix. A summary of the features added to NOTRUMP, which comprises the AP600 version (notrump-ap600), is as follows:

- SIMARC (SIMulator Advanced Real-time Code) drift flux methodology implementation
- General drift flux model modifications
  - Modified Yeh drift flux correlation for use with the SIMARC drift flux method
  - Inclusion of general droplet flow correlation when void fractions are between 0.95 and 1.0 when using the improved TRAC-PF1 flow regime map
  - Modification of the bubbly and slug flow distribution parameter (C_0)
- Use of a net volumetric flow-based momentum equation
- Implementation of the EPRI/Flooding vertical drift flux model
- Modifications to allow overriding of the default NOTRUMP contact coefficient terms for formation of regions

- Implementation of internally calculated liquid reflux flow links
- Implementation of the Mixture Level Overshoot model
- Modified Bubble Rise/Droplet Fall model logic
- Activation of the simplified pump model
- Implicit Fluid Node Gravitational Head model implementation
- Horizontal Levelizing model implementation
- Revised Unchoking model implementation
- Implementation of a revised Condensation heat link model
- Implementation of Zuber Critical Heat Flux model
- Revised Two-Phase Friction Multiplier logic
- Addition of the Henry-Fauske/HEM Critical Flow Correlation
- Improved Flux Node Stacking model logic
- Revised iteration method for Transition Boiling Correlation in metal node heat links

NOTRUMP was validated against the AP600 test data that includes all the unique features of the AP600 passive safety system design. This validation includes the Automatic Depressurization System (ADS), Core Makeup Tank (CMT), and integrated system response from SPES-2 and OSU. The AP600 Code Applicability Document (Reference 4) discusses NOTRUMP and its application to the AP600 SBLOCA analysis, providing the basis for NRC review of NOTRUMP for the AP600 design. The purpose for the integral systems tests was to provide the database to cover the range of applicability for NOTRUMP, as well as other codes.

The NOTRUMP code was compared to the separate effects AP600 test results and both integral systems tests. The process of comparing the code to the data is shown in Figure 3-1, in which the specific correlations in the code were compared to the separate effects tests while the code, as a whole, was compared to the integral systems tests. Figure 3-2 shows the relationship between the separate effects tests and the integral systems tests for the NOTRUMP code.

Using the integral test results as a guide, the separate effect tests and/or the literature were used to improve particular models or correlations. The resulting improved code, with revised correlations, was then compared to the integral systems test results, as shown in Figure 3-1. The detailed documentation associated with the NOTRUMP validation effort can be found in Reference 3 and subsequently resulted in the issuance of the NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600

Standard Design" (Reference 5) by the USNRC. This FSER applies to Version 35.0 of the NOTRUMP code utilized for AP600 applications (i.e., notrump-ap600).

For application to the AP1000 plant design, the same NOTRUMP computer code, as approved for AP600 analyses, is used with code corrections, as reported and assessed in the annual 10CFR50.46 reporting letters (References 6–8), and additional user convenience features being implemented. Summaries of the corrections performed and their impact on the AP600 analyses were determined by performing an AP600 specific calculation (see Appendix C for additional details). A summary of the impacts are as follows:

- Correction to a coding error for the implicit treatment of gravity head in NOTRUMP continuous contact flow links. This correction was deemed to have a negligible impact on the AP600 plant response.

- Correction for an error discovered in the implementation of certain droplet fall models in NOTRUMP. This correction was deemed to have a negligible impact on the AP600 plant response.

- Inconsistent updating of certain mass and volumetric rate variables during portions of the SBLOCA transient. Based on the impacts observed on traditional PWR designs, it is expected that this error correction will have a negligible impact on the AP600/AP1000 analysis results as well.

Errors were discovered in the AP600 NOTRUMP code following the termination of code error tracking (subsequent to the release of NOTRUMP Version 37.0). These errors, while corrected in the standard NOTRUMP Evaluation Model (Version 38.0), were not implemented in the code utilized in the scoping analysis (Reference 9). Since an AP600 specific assessment of these errors was not available, the estimated impact on the AP600 design was determined based on the results from traditional PWR simulations (See Appendix C for additional details). These errors are corrected in the NOTRUMP code for the AP1000 analysis. The errors and their expected impact on AP600/AP1000 analyses are as follows:

- Correction to mixture level tracking/region depletion model errors. A majority of this correction involved the implementation of the AP600 developed mixture level tracking model into the standard Evaluation Model; however, an improvement (non-error correction) was performed, which would impact the AP600 version of the code as well. The correction involves the treatment of metal node properties when fluid nodes, to which the metal nodes are connected, have depleted their inventory in a given time step. Due to the nature of the AP600/AP1000 SBLOCA transient, this change is expected to have a negligible impact on results.

To confirm the conclusion reached regarding the impact of the region depletion model correction on the AP600 design, an AP600 specific simulation will be performed with the corrected code version. However, as shown in Appendix C, implementation of this correction did not have a significant impact on conventional Westinghouse PWRs, and it is not expected to have a significant impact on the passive plant analysis.

3.2 CODE ACCEPTABILITY AP600

The following sections present the basis for the acceptability of the NOTRUMP code to the AP600 plant design as excerpted from the AP600 FSER (NUREG-1512, Reference 5). The italicized text is excerpted directly from the AP600 FSER.

3.2.1 Code Acceptability Basis – FSER

Westinghouse performed SBLOCA analyses using the NOTRUMP code as documented in WCAP-14206 (Reference 4) and WCAP-14807 (Reference 3). NOTRUMP was assessed as a 10CFR50.46, Appendix K, evaluation model. The acceptability of NOTRUMP for AP600 application was documented in NUREG-1512 (Reference 5). This acceptability was based on the review of the AP600 SBLOCA analytical results, Phenomena Identification and Ranking Table, analytical models, component models, code qualification, regulatory compliance, and ACRS review.

3.3 ISSUES FOR AP600 NOTRUMP – FSER

During the review process associated with the NOTRUMP code, issues were identified during the generation of NUREG-1512 (Reference 5). The following details the issues identified and the method utilized to address them such that a conservative calculation results. This information is excerpted from the NOTRUMP Final Validation Report (Reference 3), Section 1.17. This represents a consolidation of the issues raised by both the NRC and ACRS as agreed to by Westinghouse. These issues will be specifically addressed as part of the AP1000 program.

The definitions used for quantification are as follows (As excerpted from Section 1.5 of Reference 3):

- EXCELLENT – The calculation lies within the data uncertainty band at all times during the transient phase of interest. This is interpreted that the code had no deficiencies that are significant. No action is required for this level of agreement.

- REASONABLE – The calculation sometime lies within the data uncertainty bands and shows the same trends as the data. This is interpreted that the code deficiencies are minor. Minor actions and/or discussions are used to explain differences.

- MINIMAL – Major data trends and phenomena are not predicted. The code has significant deficiencies, and incorrect conclusions may be drawn based on the calculations without the benefit of data. If the deviation of the code calculations is known, then the minimal agreement may be acceptable for lower-ranked items in the PIRT.

- INADEQUATE – Modeling the phenomena is beyond the capability of the code. The questions then becomes how important are these phenomena for describing the transient and having confidence in the results and their application to the plant.

ADS-4: Two-Phase Pressure Drop

The assessment results were deemed to be minimal due to the lack of momentum flux terms, which resulted in the underprediction of two-phase pressure drop during noncritical flow conditions. The utilization of upper bound loss coefficients in this flow path and the application of a 6-foot IRWST water level penalty treated this deficiency in the AP600 analyses. This treatment results in a conservative prediction of IRWST injection, which is the long-term cooling source for the AP600 design.

Downcomer Mixture Level

For the double-ended direct vessel injection (DEDVI) simulation, the downcomer mixture level was deemed to be minimal due to the fact that NOTRUMP code is a one-dimensional code and the DEDVI transient is two-dimensional during the early portions of the transient. The application of the IRWST level penalty and the use of a range of discharge coefficients (Cd) were utilized to account for this deficiency for this break simulation.

Phase Separation at Tees

The phase separation at Tee junctions in the cold legs was deemed to be conservative in that the treatment in the NOTRUMP code results in artificial balance line refilling which causes a delay in CMT draining and subsequent ADS system actuation. No change to the model was required due to its conservative nature.

The phase separation at Tee junctions in the hot legs was deemed to be minimal due to the use of an ad-hoc model. The impact was deemed to be small as the liquid flow out of the ADS-4 paths are controlled by constant system inventory and are thus self-correcting. The application of the IRWST level penalty was used to conservatively bound the expected impact.

Pressurizer and Surge Line CCFL

This model was assessed as minimal, but conservative, provided the vapor flow to the component was correct. This apparent weakness was caused by low vapor flow to this component resulting from low pressure drop through the ADS-4 paths when noncritical flow was predicted to occur.

Pressurizer and Surge Line Level Swell

This model was assessed as minimal, and nonconservative, during the pressurizer drain period following ADS-4 actuation. This was caused by the poor ADS-4 pressure drop prediction, which was confirmed by studies with increased ADS-4 resistance. For the AP600 application, this deficiency was compensated for by the application of the 6-foot IRWST level penalty that delays IRWST injection.

PRHR Heat Transfer/Recirculation Flow

These areas were deemed to be minimal, but conservative, provided the primary flow through the PRHR is low. Westinghouse committed to confirm that the flow velocity through the PRHR primary tubes would be less than 1.5 ft/sec in all AP600 simulations. In addition, the PRHR is removed from the model

following ADS 1-3 actuation to further reduce the depressurization rate. Should the flow rate through the PRHR be higher than 1.5 ft/sec for any significant period of time, the calculation for the limiting case (minimum mass or highest PCT) would be repeated with the PRHR heat transfer surface area reduced by 50 percent to account for the potential heat transfer overprediction.

Noncondensable Gas Injection

Since the AP600 NOTRUMP code does not contain a noncondensable gas model, it can not accurately predict the plant behavior as a result of the introduction of noncondensable gasses from the Accumulators. To assure conservatism in accounting for this deficiency, the primary heat removal system in NOTRUMP (i.e., PRHR Heat Exchanger) will be removed from the model prior to Accumulator empty. This conservatively bounds the effect of the introduction of noncondensable gases into the PRHR heat exchanger. It was determined that the accumulation of noncondensable gases into other model locations such as steam generator tubes and the CMTs would not adversely impact plant performance.

3.4 NOTRUMP CODE ACCEPTABILITY FOR AP1000

This section contains a review of the pertinent information associated with the application of the NOTRUMP code, as approved for AP600, to the AP1000 plant design. It provides a review of the PIRT issues, phenomenological issues, scaling issues, and margin issues as well as addressing the issues identified from the AP600 program.

3.4.1 PIRT Issues

A review of the PIRT was performed in Section 2.0 of the AP1000 PIRT and Scaling Assessment report (Reference 10) and concluded the following related to important SBLOCA phenomena:

- ADS-4 subsonic, two-phase flow should be raised to a high importance.
- Upper plenum/hot leg entrainment during the post-ADS period should be raised to a high importance level.
- Pressurizer surge line countercurrent flow/flooding during the ADS-IRWST period should be raised to a high importance level.

The above items are not really new phenomena but rather the change in rankings is a result of the lessons learned from the AP600 test and analysis program. The issues identified above apply to both the AP600 and AP1000 designs and do not constitute new issues. These issues were previously reviewed by the ACRS/NRC during the review of the NOTRUMP application to the AP600 plant design.

3.4.2 Phenomena Issues

As a result of the scoping analyses performed in WCAP-15612 (Reference 9), no new phenomena were observed.

3.4.3 Scaling Issues

As a result of the efforts performed in the AP1000 PIRT and Scaling Assessment report (WCAP-15613, Reference 10), it was concluded that the AP600 test program can be successfully applied to support code validation for application to AP1000 safety analysis; subsequently, tests that are better scaled to the AP1000 have been performed and used for code validation as discussed in subsection 3.5.3. In addition, it was also stated that "For small break LOCA events, computer codes that acceptably predict SPES-2 and OSU behavior can be used to conservatively analyze the performance of the AP1000. Moreover, codes that predict the high-pressure phases of the transient (i.e., prior to ADS-4 actuation) will acceptably predict the high-pressure portion of the SBLOCA transient for the AP1000 plant. Codes that predict the lower pressure phases (i.e., post ADS-4) will acceptably predict the performance of the AP1000 for the low pressure phases of the SBLOCA transient."

The NOTRUMP code has been validated against both the OSU and SPES-2 integral test facilities (Reference 3) and deemed to provide reasonable predictions of the highly ranked PIRT phenomena, as described in Section 3.2.1.5 of this document. As such, the NOTRUMP code can be utilized for the prediction of SBLOCA phenomena anticipated in the AP1000 plant design. Additionally, the scoping analyses performed in Reference 9 indicate no new phenomena with comparable safety margins to those observed for the AP600 plant design.

3.4.4 Margin Issues

As a result of the SBLOCA scoping analyses performed in WCAP-15612 (Reference 9), the AP1000 plant performance was observed to exhibit safety margins comparable to that observed for the AP600 plant design. In fact, due to the component size increases associated with the AP1000 design, the breaks analyzed respond like smaller breaks in the AP600 plant design. As a result, comparable break sizes respond in a more benign fashion than observed for the AP600 plant design. A break spectrum has been performed and reported in the AP1000 DCD. The expectations regarding available margins for the AP1000 plant design obtained via the scoping studies have subsequently been confirmed.

3.4.5 How Issues Are Addressed for AP1000

The approach used to address the code issues identified as part of the AP600 design is as follows:

1. Start with the computer codes as approved for passive plant analysis in the AP600 design certification program.

2. Confirm the adequacy of the codes for analysis of the AP1000 design.

3. Address potential concerns identified as a result of the AP600 design certification review.

4. Reach a consensus regarding the acceptability of the methods utilized.

The confirmation of the adequacy of the computer codes for analysis of the AP1000 design is addressed via the following steps:

1. Identification of important phenomena (via PIRTs) that must be addressed by the code. (Completed via the submittal of the AP1000 PIRT and Scaling Report, Reference 10)

2. Identification of correlations and model used in the code to address important phenomena. (Completed via the AP600 Design Certification Program, References 3 and 5)

3. Demonstration of the existence of an adequate test data base to support validation of the models/correlations via scaling analyses. (Completed via the submittal of the AP1000 PIRT and Scaling Report, Reference 10)

4. Demonstration that the limitations identified in the AP600 FSER are adequately addressed for the AP1000 program. (Addressed in this report.)

Of the items listed above, only the approach to address the limitations identified during the AP600 review have yet to be performed. To address code limitations, one of the following approach(s) may be used:

1. Performance of plant design modification to increase available margin.

2. Performance of additional validation efforts with the computer codes versus appropriate test(s).

3. Performance of an evaluation of the available plant margin.

4. Performance of supplementary analyses using appropriate means (e.g., alternate code simulations).

5. Performance of code/model enhancements to address the identified deficiencies.

6. No change required if the model is deemed to result in a conservative calculation.

The following discussion addresses how the issues identified in the AP600 program (and discussed in Section 3.3) are addressed for the AP1000 program.

For application to the AP1000 program, the validation program developed/analyzed for the AP600 is used as supported by the work performed in Reference 10. Note that areas identified/assessed as being "minimal" in terms of acceptability, per Section 1.17 of Reference 3 and as stated in Section 3.2.2, are evaluated for adequacy in the AP1000 program. A summary of the assessment items are provided in Table 3-1. Of these items, the areas that need to be addressed via the criterion defined above, in the sequence presented in Section 3.3, are as follows:

ADS-4: Two-Phase Pressure Drop

The method used to address this item is the use of a previously evaluated modeling modification (i.e., ADS-4 resistance increase based on the results of a stand-alone detailed momentum flux model).

The need for momentum flux terms to accurately model the ADS flow paths (particularly ADS-4 during the sub-sonic flow period) will result in the need to improve this modeling. This was an area previously deemed to be inadequate in the AP600 test and analysis program and required the implementation of penalties (IRWST Level reduction) to compensate for this deficiency. For the AP1000 program, this deficiency will be addressed via the implementation of an ADS-4 resistance increase at the time when the ADS-4 flow paths transition to unchoked flow conditions. This methodology was demonstrated on AP600 analyses to be similar in nature to the imposition of the IRWST level penalty while more directly addressing the NOTRUMP code deficiency (i.e., lack of a detailed momentum flux model in the ADS-4 flow paths). The ADS-4 resistance increase is developed in the same fashion as utilized in response to the AP600 Request for Additional Information (RAI 440.796F, Part a). Specifically, a detailed stand-alone momentum flux model (called FLOAD4) has been developed for the AP1000 ADS-4 specific flow geometry. The results of the FLOAD4 model were then used to generate an effective ADS-4 resistance increase to be implemented into the NOTRUMP model at the time when the ADS-4 flow paths transition to unchoked flow. Figure 3-8 gives FLOAD4 calculated results for the AP1000 ADS-4 flow path on the non-pressurizer loop with both squib valves open. Figures 3-9 through 3-24 give FLOAD4 calculated results for static pressure in the two branch lines as a function of distance from the hot leg for various hot leg pressures and flow qualities.

An independent calculation was performed for 100-percent steam flow using Crane Technical Paper 410 methods. Crane lists the following equation (eq. 3-20) for compressible fluids:

$$w = 0.525\, Y\, d^2\, (DP / K\, v)^{0.5}$$

where:

w is the mass flow (lb/sec)

Y is an expansion factor from Crane page A-22 for k = 1.3 (water, steam)

d is the pipe inside diameter (used to calculate K)

DP is the differential pressure (psi)

K is the resistance coefficient, velocity head loss

v is the upstream mixture specific volume (ft^3/lb)

As an example, for the AP1000 with a hot leg pressure of 20 psia and a containment pressure of 14.7 psia:

Y = 0.87 from Crane page A-22 with the K shown below and a DP/P = 0.26

d = 15.82 in. (equivalent inside diameter of two ADS-4 branch lines, 14" sch 160)

DP = 5.3 psi (for listed RCS/containment pressure)

K = 3.81 (ADS-4 line B with both valves open)

v = 20.09 ft^3/lb (saturated steam at 20 psia)

w = 30.08 lb/sec

The results of this calculation are plotted against the FLOAD4 data in Figure 3-8. The FLOAD4 result with 100-percent steam compares well with this calculation using Crane compressible flow equation.

Use of this method more accurately reflects the ADS flow distributions and ultimately the onset of IRWST injection flow. For the AP1000 plant DCD analyses, the FLOAD4 model developed for the AP1000 plant design resulted in the required resistance increase determined to be [ ]^{a,c}.

Despite the fact that a mechanistic approach is used by FLOAD4 in modeling the two-phase pressure drops, the validation of the FLOAD4 model against applicable two-phase pressure drops data was deemed necessary so that the FLOAD4 solution can be used as the benchmark for the NOTRUMP calculation.

The total pressure drop is determined by adding the following terms:

- Gravity contribution
- Acceleration contribution (due to area contractions and two-phase mixture compressibility)
- Frictional term (irreversible losses because of wall drag, interfacial drag, and irreversible losses at the fittings; that is, valves, tees, and area contractions)

With the exception of low-flow quality, the gravity term is negligible. A simple homogenous approach can be used to determine its contribution. Since the expected contribution is small relative to the other terms, a certain degree of uncertainty in its value is acceptable.

More important is the effect of the acceleration terms and irreversible losses. In the following, the acceleration terms and irreversible losses are determined using different two-phase pressure drop models. The models are compared to applicable two-phase pressure drop data. It is found that a homogeneous approach for the calculation of two-phase pressure drops is typically acceptable.

Distributed Frictional Losses

Experimental pressure drop for two-phase flow in rectilinear ducts of any inclination was collected and structured in a Data Bank called MIDA at the "Politecnico di Milano" (Reference 14). The MIDA database includes about 24,000 two-phase pressure drop data points. The CESNEF-2 correlation was developed from the MIDA database. The authors reported that CESNEF-2 has an average error of 8.9 percent and 73 percent of the data points fall within +20 percent.

In the CESNEF-2 correlation, the total pressure drops are evaluated based on a mechanical energy equation where the total pressure drops are given by:

$$-dp = dp_g + dp_a + dp_f$$

The gravitational term dp_g and the acceleration term dp_a are calculated assuming a homogeneous model. The frictional term is calculated with a detailed model that accounts for slip between the phases. The total pressure drop is correlated to the data such that any error in the evaluation of the gravitational term and acceleration term is "compensated" in the frictional term.

The predictions of the CESNEF correlation were compared to a subset of its database, to further support its validity for the AP1000 conditions (low-pressure and high-flow quality). Figure 3-25 shows the predicted distributed pressure drops versus the measured pressure drops (kPa/m) for a subset of tube data at low pressure (< 0.6 MPa). Figure 3-26 shows a similar comparison by selecting data points at high quality (X > 0.7).

Figures 3-25 and 3-26 mostly show that the CESNEF-2 predictions are in good agreement with applicable test data (mostly within ±20 percent). Therefore, CESNEF-2 can be used to predict the distributed pressure drops in the ADS-4 line for the AP1000.

The prediction of CESNEF-2 was compared to predictions obtained from homogeneous models. Figure 3-27 plots the predicted distributed pressure drops, assuming the typical geometry of ADS-4 piping (14 inch Sch. 160 pipe), for the range of flow quality of interest for the AP1000. CESNEF-2 is compared to two homogeneous models. The first is based on using the friction factor reported in Crane, which accounts for the roughness of the pipe and calculates the pressure drop considering a homogenous mixture (mixture density and mixture viscosity). For a fully turbulent flow in the pipe geometry and roughness of interest, the friction factor obtained from Crane is f = 0.01314 (Reference 16). The second is another homogenous model, which assumes a smooth pipe with the friction factor calculated from the Blasius equation.

Figure 3-27 confirms that the homogeneous model provides reasonable results, which are in agreement with the CESNEF-2 data.

Irreversible Losses at Fittings

The irreversible losses at the fittings (valves, tees, and contractions) are calculated assuming the equivalent L/D approach, which is typical in single-phase applications. This is a reasonable approach since the two-phase mixture is homogeneous, especially at high-flow quality ratios.

Acceleration Contribution at Sudden Contractions

The acceleration term is calculated based on the homogenous model. This approach was confirmed by comparing the predicted acceleration term to applicable data for contractions.

The ADS-4 flow paths in the AP1000 contain significant sudden contraction area changes where the flow accelerates in the ADS-4 piping and results in notable pressure drop in the core exit region flow path. Experience from single-phase flow indicates that pressure drop through a sudden contraction is largely due to a reversible acceleration pressure drop and an irreversible viscous loss due to momentum transfer in the downstream section. The largest area change in the AP600 and AP1000 ADS-4 flow paths occurs at the entrance to the ADS-4 flow path from the hot leg. The area ratio of the AP600 ADS-4 off-take to the hot leg (A_{ads4}/A_{hleg}), for example, is about 0.11 and for the AP1000 is about 0.22.

It is desired to show that the pressure drop associated with sudden contractions can be acceptably predicted by comparison with single-phase and two-phase sudden contraction data (see the pressure drop data sources section following). For two-phase conditions, homogeneous-based methods seem to provide acceptable prediction of pressure drop through sudden contractions; whereas, attempts using

more sophisticated separated flow based models have offered little or no improvement over homogeneous-based models. It has been postulated that this may be the case as the two-phase flow, regardless of the flow regime upstream of the sudden contraction, is forced to mix (liquid and vapor phases exchange momentum) or homogenize as it encounters the flow contraction. The following discussion presents pressure drop predictions of sudden contraction data for two-phase and single-phase flow.

Three sources of pressure drop data are presented:

- Geiger and Rohrer, "Sudden Contraction Losses in Two-Phase Flow," Journal of Heat Transfer, 1966.
- Schmidt and Friedel, "Two-Phase Pressure Drop Across Sudden Contractions in Duct Areas," Int. J. Multiphase Flow, 1997.
- Ferrell and McGee, "Two-Phase Flow Through Abrupt Expansions and Contractions," 1966.

Comparison is made to two homogeneous methods:

- Homogeneous Method Using Mechanical Energy Equation

$$\Delta p_{\substack{\text{sudden}\\ \text{contraction}}} = \frac{G_m^2}{2\rho_m}[1-\sigma^2 + K_{1\phi\ \text{contraction}}]$$

where ρ_m is homogeneous mixture density and σ is area ratio of sudden contraction.

- Homogeneous Method Using Momentum Equation

$$\Delta p_{\substack{\text{sudden}\\ \text{contraction}}} = \frac{G_m^2}{\rho_m}[1-\sigma]$$

Results applying the homogeneous methods are shown in Figures 3-28 through 3-33. Figure 3-34 shows comparison against single-phase data. The homogeneous mechanical energy method generally predicts the two-phase data reasonably well. The homogeneous momentum method provides a conservative prediction of the two-phase data.

Application of CESNEF-2 to Entire ADS-4 Piping Network and Comparison to FLOAD4 Predictions

Based on the previous discussion, the CESNEF-2 correlation was used to calculate the pressure drops in the ADS-4 piping (Figure 3-35). Note that the gravitational term and acceleration term are calculated assuming a homogenous model. This is consistent with the use of CESNEF-2 as discussed in Reference 15. Also, the comparison to data in the previous section indicates that a homogenous approach in the determination of the acceleration at the contractions is reasonable. This calculation also accounts

for the acceleration term due to the expansion of the two-phase mixture along the system. However, it was found that this contribution is small compared to the acceleration contribution at the area contractions.

The pressure drops predicted using the CESNEF-2 model were compared to the FLOAD4 results. The pressure distribution predicted using the FLOAD4 model is shown in Figure 3-24. The figure shows the calculation for the case where the inlet pressure (hot leg) is 22 psia and the flow quality is 100 percent (pure steam).

The initial pressure drop is due to the acceleration at the inlet of the ADS-4 stand pipe. A pressure recovery is calculated across the tee because of the flow split and deceleration. The pressure decreases and recovers across the gate valve (at about 30 feet). The pressure drop at the end is due to the acceleration at the inlet of the squib valve. There is no pressure recovery from the restricted area in the squib valve to the containment (14.7 psia). The kinetic energy at the throat is assumed to be fully dissipated in the expansion to the containment. Note that the total flow is a result in the FLOAD4 calculation because the pressure at both the inlet and the outlet of the system is provided as input.

Figure 3-36 shows the calculated vapor flow as a function of the inlet pressure and for several values of flow quality. Each square point in the plot is from the FLOAD4 calculation at 10-percent flow quality increments, from 10 to 100 percent, increasing from the bottom to the top. Also shown on Figure 3-36 are the results obtained from the CESNEF-2 calculations for two points. Figure 3-36 shows that FLOAD4 is in agreement with the CESNEF-2 data.

In NOTRUMP, the calculation of the ADS-4 total pressure drop was adjusted to compensate for the missing momentum flux. Then NOTRUMP results were benchmarked against the FLOAD4 model. Figure 3-37 shows the comparison between the integrated vapor flow predicted by NOTRUMP and FLOAD4 for the same boundary conditions. The figure shows that the agreement is good with NOTRUMP and shows that NOTRUMP tends to provide conservative results.

Downcomer Mixture Level

The method utilized to address this item is the use of a previously evaluated modeling modification (i.e., ADS-4 resistance increase based on the results of a stand-alone detailed momentum flux model and break discharge coefficient study).

While not specifically addressing the multi-dimensional aspects of the downcomer behavior that results from this break location, the modifications imposed assure conservative behavior prior to the onset of IRWST injection, which terminates the inventory depletion period. As observed during the code validation, the discrepancy in downcomer behavior is resolved by the time ADS 1-3 blowdown is completed as evidenced by the good agreement between the test and NOTRUMP predictions for both SPES and OSU. It is also noted that this misprediction in downcomer behavior does not adversely impact core mixture level.

For the DEDVI line simulation, the downcomer mixture level was deemed to be minimal due to the one-dimensional nature of the NOTRUMP code and the two-dimensional nature of the DEDVI transient

during the early portions of the transient. This was addressed via the implementation of the IRWST level penalty and the performance of a range of break discharge coefficients for the AP600 program.

Application of the ADS-4 resistance increase and a range of discharge coefficients (C_d) is applied to the DEDVI line break for the AP1000 program to assure the limiting break size has been captured.

Phase Separation at Tees

The phase separation at Tee junctions in the cold legs was deemed to be conservative in that it resulted in delayed draining of the CMT and subsequent ADS system actuation. No change in the model was required due to its conservative nature. This model will remain unchanged in application to the AP1000.

The phase separation at Tee junctions in the hot legs connected to the ADS-4 paths was deemed to be minimal due to the use of an ad-hoc model. Entrainment/phase separation can impact the flow quality encountered at the ADS-4 discharge valves and affect the capability of the plant to achieve stable IRWST injection flow. The use of the ad-hoc model to account for the effects of entrainment/phase separation was utilized in the analysis of the AP600 and integral test facilities and was determined to have a negligible impact on plant results. Further justification for the NOTRUMP model is provided by the sensitivity study to hot-leg/upper plenum entrainment discussed in Appendix F. In particular, the study shows that SBLOCA behavior is relatively insensitive to hot-leg/upper plenum entrainment and, therefore, also to the degree of phase separation at the hot-leg tee junction to ADS-4.

Pressurizer and Surge Line CCFL

This model was deemed to be minimal but conservative for AP600 provided the vapor flow to this region was correct.

Due to deficiencies in the ADS-4 flow path modeling, early IRWST injection relative to the OSU integral test data was thought to be related to pressurizer draining, particularly surge line flooding. However, as shown in ADS-4 resistance increase studies performed with the NOTRUMP code for AP600, pressurizer draining and IRWST injection initiation times more closely match the behavior observed in the test data. The increases in ADS-4 resistance were implemented to account for the lack of a detailed momentum flux model in the NOTRUMP code. The resistance increases utilized were based on the results of a detailed stand-alone momentum flux model of the ADS-4 flow paths as discussed in the response to AP600 RAI 440.796, Part a. As such, with the implementation of the ADS-4 resistance increases, it is expected that the pressurizer drain behavior is conservatively captured for AP1000 and no additional modification to this model is required.

Pressurizer and Surge Line Level Swell

The methods utilized to address this item are the use of a previously evaluated modeling modification (i.e., ADS-4 resistance increase based on the results of a stand-alone detailed momentum flux model) and the performance of a supplementary analysis utilizing the WCOBRA/TRAC-AP code.

This model was assessed as minimal, and non-conservative, for AP600 during the pressurize drain period following ADS-4 actuation. This was caused by the poor prediction of the ADS-4 pressure drop, which

was confirmed by studies with increased ADS-4 resistance. The poor prediction of ADS-4 pressure drop results in the core vapor being preferentially discharged through the ADS-4 locations. As a result, the vapor flow entering the pressurizer component is low resulting in the underprediction of CCFL in the pressurizer surge line and the pressurizer drains more rapidly then observed in the test. As observed in Figures 3-3 and 3-4, the increase in ADS-4 resistance more accurately reflects the behavior observed in the test. For the AP1000 application, the application of the ADS-4 resistance increase corrects this behavior for the reasons stated previously.

PRHR Heat Transfer/Recirculation Flow

These areas were deemed to be minimal, but conservative, for AP600 if primary flow through the PRHR was low. As such, the methodology associated with the confirmation of the PRHR heat exchanger flow velocities and implementation of heat transfer modifications, as discussed in Section 3.3 will be followed, as necessary, to assure conservatism.

The flow velocity for the AP1000 design may exceed 1.5 ft/sec for much of the time during an SBLOCA event. The NRC staff, therefore, requires that Westinghouse define and justify what is considered to be a "significant period of time" to trigger a reduction in PRHR surface area and to justify that a 50-percent reduction of heat transfer area is conservative given comparisons with data appropriate for the AP1000 design.

As stated in section 15.6 of the AP1000 DCD, the small break LOCA analysis performed for AP1000 that is presented in Chapter 15 of the DCD uses the heat transfer penalty on PRHR heat transfer that was identified for the AP600, for cases when the velocity in the PRHR tubes is greater than 1.5 ft/sec. For AP1000, this penalty was applied for the entire transient, regardless of the velocity in the PRHR tubes. The following provides our justification for this penalty.

The Thom correlation in NOTRUMP slightly overpredicts the heat transfer relative to the modified Rosenhow correlation that was developed from the AP600 PRHR test data by 6 to 8 percent depending on primary side inlet conditions. Reducing the heat transfer area by 50 percent and using the Thom correlation results in a reduction in the heat transfer relative to the modified Rosenhow correlation of 11 to 13 percent for the same conditions. See Appendix D for additional details.

Therefore, the penalty on heat transfer for the PRHR as applied to the AP1000 SBLOCA analysis is conservative. In addition, the PRHR model is removed prior to ADS-4 actuation as done for AP600.

Noncondensable Gas Injection

The removal of the PRHR model prior to the introduction of noncondensable gases conservatively bounds the expected behavior.

The methods used in the SBLOCA analyses to account for noncondensable gas introduction (i.e., PRHR removal prior to accumulator empty time) are used on the AP1000 design as well.

Transition Boiling Model Related

Per the AP600 FSER issued by the NRC (Reference 3), the use of the transition boiling correlation, for fuel rod heat transfer, was not specifically reviewed by the NRC as part of the AP600 program. This model is unchanged from the standard NOTRUMP Evaluation Model as documented in Reference 2. Since the correlation being utilized is standard in many Westinghouse analytical tools, its range of applicability to the AP600/AP1000 operating conditions could be confirmed should core uncovery be observed, which is not the case for AP1000 breaks below 10 inches. For the 10-inch break, a conservative heatup model is used for the brief period of uncovery.

3.4.6 Additional NOTRUMP Considerations for AP1000

For the AP600 and AP1000, SBLOCA events are not the most limiting events with regard to calculated PCT. However, this category of events is the most challenging with respect to the integrated performance of the passive core cooling system features, such as automatic depressurization and gravity injection. The AP600 test and analysis programs showed that the transition from ADS depressurization to IRWST injection during the SBLOCA is of greatest concern as minimum reactor vessel inventory typically occurs during this transition phase. Consequently, the pivotal SBLOCA-related issue identified during the AP600 Design Certification review with the ACRS became the ability of the NOTRUMP code to conservatively predict the onset of IRWST injection following actuation of the ADS, as gravity injection is critical in providing long-term recovery of reactor vessel inventory. Therefore, the primary means of resolution for this issue was to demonstrate that the NOTRUMP code could conservatively predict the onset of IRWST injection in the AP600 integral effects tests, which were shown to be adequately scaled to the AP600 during this transition phase. As the AP1000 has also been shown to be adequately scaled to the AP600 integral effects tests (Reference 3), the same means of resolution is used for AP1000 except instead of conservative treatment of the IRWST gravity head (level penalty), conservative treatment of the ADS-4 resistance is applied as described below.

The reason for using an ADS-4 resistance based adjustment to ensure conservative prediction of the onset of IRWST injection in NOTRUMP for AP1000 as opposed to using an IRWST level penalty as in AP600 is that there is little uncertainty associated with single-phase gravity injection from the IRWST. The gravity head and single-phase hydraulic resistance are well known and understood. However, the onset of IRWST injection is also very dependent upon the backpressure in the reactor vessel (downcomer). Reactor vessel pressure is in turn controlled by the venting of steam through the ADS outlet paths. Steam venting through the ADS paths is strongly influenced by complex, two-phase flow interactions in the hot legs and ADS piping involving entrainment and two-phase pressure drop through the ADS valves and piping including momentum flux. These phenomena have a much higher uncertainty, are not as well understood, and, in general, are not accurately predicted by two-phase thermal-hydraulic codes. Therefore, any adjustment to the analysis code model should be to the two-phase pressure drop associated with the ADS-4 vent paths.

NOTRUMP ADS-4 Resistance Increase Effect

In order to demonstrate the effect of the ADS-4 resistance adjustment on the NOTRUMP results, the information generated in support of the May 11th and 12th 1998 ACRS Thermal Hydraulic subcommittee meeting is summarized below.

As presented in the response to AP600 RAI 440.796F, Part a, the detailed momentum flux model, developed for the OSU facility, calculated a required resistance increase of approximately 35 percent would be necessary to account for the model deficiency in the NOTRUMP OSU model. NOTRUMP simulations for the OSU facility were performed in which a resistance increase of 42 percent was applied to the ADS-4 flow paths at the transition to non-critical flow conditions. The 42 percent resistance was available from a series of sensitivity studies performed with the NOTRUMP OSU model. While the value is not an exact match to the required increase, as calculated by the detailed stand-alone momentum flux model, it provides an estimate of the impact of the model adjustment. Figures 3-3 through 3-7 present comparisons of the OSU 2-inch cold leg break simulation (Test SB18) between the test data, the base NOTRUMP model used in Final Validation Report (Reference 3), and the adjusted NOTRUMP model results generated in support of the ACRS Thermal Hydraulic subcommittee meeting. As can be seen by these figures, the pressurizer drain behavior (Figures 3-3 and 3-4), ADS-4 integrated flow behavior (Figure 3-5), and IRWST-1 injection flow (Figures 3-6 and 3-7) are more accurately reflected by the adjusted NOTRUMP model and result in a conservative prediction of the IRWST injection flows. This demonstrates that the major contributor to the deviations between the NOTRUMP model and the test data results from the deficiency in the ADS-4 pressure drop during the non-critical flow period. It also demonstrates that this is a more direct means of adjusting the NOTRUMP model in lieu of the originally utilized IRWST level penalty adjustment.

3.5 ADDITIONAL ACRS ISSUES FOR AP1000

As a result of meetings held among Westinghouse, the USNRC, and the ACRS on the AP1000 plant design, additional verification/simulation items were requested. The items addressed in this section are those associated with the following topics:

- Hot leg/upper plenum entrainment
- Assessment of level swell phenomena
- Supplemental NOTRUMP simulations of OSU APEX-AP1000

3.5.1 HOT LEG/UPPER PLENUM ENTRAINMENT

Supplemental NOTRUMP sensitivity studies were performed as described in detail in Appendix F of this document. The results indicate that the AP1000 plant design is relatively insensitive to hot leg/upper plenum entrainment effects and adequate inventory is available to maintain core cooling.

3.5.2 ASSESSMENT OF LEVEL SWELL PHENOMENA

Additional validation of the NOTRUMP core model has been performed for the time period of interest; namely, the ADS-4 to IRWST transition phase. The results indicate that the NOTRUMP level swell model, which uses the Cunningham-Yeh correlation, is in good agreement with the data for the range of conditions applicable to the AP1000 plant. The details regarding the test data assessment are found in Appendix G of this document.

3.5.3 SUPPLEMENTAL NOTRUMP SIMULATIONS OF OSU APEX-AP1000

To further confirm the applicability of the NOTRUMP computer code to predict the AP1000 plant behavior for SBLOCAs, the revised OSU APEX test facility (References 12 and 13) was modeled with the Advanced Plant version of the NOTRUMP computer code. The details regarding the results of the simulations performed are found in Appendix E of this document. The results obtained support the acceptability of the NOTRUMP code for analysis of the AP1000 plant design.

3.6 ASSESSMENT OF DG-1096 RELATED METHODS

In a workshop (April 9, 2001) to discuss Draft Regulatory Guide DG-1096, several attributes were discussed which should be considered in determining the extent to which the DG process should be used in the development, assessment, and application to an evaluation model. These are:

- Novelty of the evaluation model compared to the currently acceptable model.
- The complexity of the event being analyzed.
- The degree of conservatism of the evaluation model.
- Risk or safety importance of the event.

For the NOTRUMP AP1000 analysis program, these issues are addressed as follows:

- The evaluation model, which is used for the AP1000 program, is the same as that used for the AP600 program with minor error corrections and user convenience features being implemented as discussed in Section 3.1 of this document. As such, no significant changes are being made to the evaluation model as approved for AP600 applications.
- While the SBLOCA event is typically not considered to be a complex event for traditional PWRs, the nature of the AP600/AP1000 designs are such that the behavior involved (automatic depressurization to low pressure conditions) results in calculation complexities. The event and underlying methodology was thoroughly reviewed for application to the AP600 plant design. The AP1000 DCD analyses do not indicate the existence of new phenomena for the AP1000 design as compared to that observed for the AP600 design.
- The evaluation model and methodology used continues to be based on the use of Appendix-K required features. As such, the model and modeling features will result in a conservative calculation with respect to the expected plant response. In addition, the AP1000 DCD analyses show significant margin to the 10 CFR 50.46 limits for the AP1000 plant design.
- The AP1000 DCD analyses, presented in Reference 9, indicate no significant change in the margin to core uncovery. Therefore, significant margins to the 10 CFR 50.46 limits exist for this plant design.

The code being utilized in support of the AP1000 design has previously undergone a detailed review as part of the AP600 design certification process with the required aspects of DG-1096 having been met. As such, the code has been approved for use on the AP600 and is considered by Westinghouse to be applicable for use on the AP1000.

3.7 CONCLUSIONS/RECOMMENDATIONS

The SBLOCA response is evaluated for the AP1000 with an evaluation model that conforms to 10 CFR 50 Appendix K. The elements of the AP1000 SBLOCA evaluation model are the following:

- NOTRUMP computer code
- NOTRUMP homogeneous sensitivity model
- Critical heat flux assessment during accumulator injection

NOTRUMP Computer Code

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the RCS. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in the AP1000 SBLOCA calculations has been validated against applicable passive plant test data. The code has limited capability in modeling upper plenum and hot leg entrainment, and did not predict the core collapsed level during the accumulator injection phase adequately. The NOTRUMP homogeneous sensitivity model and the critical heat flux assessment during the accumulator injection phase supplement the base NOTRUMP analysis to demonstrate the adequacy of the design.

NOTRUMP Homogeneous Sensitivity Model

In order to address the uncertainties associated with upper plenum/hot leg entrainment following ADS-4 operation, a plant sensitivity study is performed with the DEDVI line break. This sensitivity maximizes the amount of upper plenum/hot leg entrainment by modeling the regions downstream of the core in a homogenous fashion. All other features described in the previous section are retained.

The methodology for the NOTRUMP homogeneous sensitivity model is described in Appendix F. This model is used to assess the DEDVI break because this is the limiting break for entrainment, and the model is used to demonstrate that even with maximum entrainment, the 10 CFR 50.46 criteria are met.

Critical Heat Flux Assessment During Accumulator Injection

An assessment is performed of the peak core heat flux with respect to the critical heat flux during the later ADS depressurization time period for a double-ended rupture of the direct vessel injection line. This time period corresponds to the accumulator injection phase of the transient. The predicted average mass flux at the core inlet and the reactor pressure from the NOTRUMP computer code base model analysis are

used as input parameters to critical heat flux correlations. NOTRUMP has been shown (Appendix E) to adequately predict mass flux and pressure for integral systems tests.

The following assessment addresses heat transfer in the core for the AP1000 DEDVI case during the time period of accumulator injection and ADS-4 blowdown. This is the equivalent time period as the 150-400 second time period in the APEX-1000 DEDVI tests in which NOTRUMP underpredicts core average void fraction.

The transient analyses performed with NOTRUMP indicated that during the accumulator injection ADS depressurization phase, core cooling is adequate. A significant amount of water is stored in the upper plenum, and the two-phase mixture level is located approximately at the center of the hot leg nozzle elevation.

In this time period, a boil-off scenario – where the mixture level drops below the top of the core, causing an almost adiabatic heatup – is prevented from occurring because the flow is more than sufficient to remove the decay heat and core exit quality is significantly less than unity. There is good flow communication between the core region and the upper plenum region as suggested by the estimated value of the Kutateladze number at the upper core plate flow restriction. This allows water in the upper plenum to periodically drain into the core while maintaining a liquid film at the rod surface.

However, a temporary temperature excursion in the region of the core would still be possible if the critical heat flux limit is exceeded. The scope of this analysis is to assess how much margin is present with respect to the critical heat flux limit during the later ADS depressurization time period, which for a DEDVI case is predicted to occur in the time window between 400 and 600 seconds (equivalent to the 150-400 second time frame in the APEX-1000 DEDVI tests) after the opening of the break.

The predicted mass flux at the core inlet is on the average constant and corresponds to 7.2 lbm ft^{-2} s^{-1} (~35 kg m^{-2} s^{-1}). The key thermal-hydraulic parameters at different times during the ADS depressurization time period are summarized in following table.

Time (sec)	Upper Plenum Pressure (kPa)	Upper Plenum Pressure (psia)	Mass Flux (kg/m^2s)	Average Heat Flux (kW/m^2)
400	1293	190	35	20.2
500	646	95	35	19.1
570	340	50	35	18.5
600	272	40	35	18.2

The above parameters are from the base NOTRUMP prediction of 7.2 lbm/ft^2-s during the ADS depressurization period for the AP1000 following a DEDVI event. During the period of interest, NOTRUMP may underpredict the void fraction due to a lack of significant condensation and thermal plume propagation in the one-dimensional downcomer model. These modeling issues result in overprediction of subcooling into the core, which lowers the void fraction in the core during this period.

In the critical heat flux assessment, the core inlet was assumed to be saturated so that the subcooling seen in NOTRUMP does not influence the critical heat flux assessment. Also, a NOTRUMP AP1000 sensitivity run was made where the core inlet was forced to saturation by increasing the downcomer condensation, and the void fraction in the core was predicted higher by using the EPRI drift flux correlation in the core. This sensitivity run was used to compare the core inlet flow relative to the base case. The core inlet flow and the system pressure were not significantly different for the two cases. This indicates that using the core inlet flow and pressure predicted in the base case DEDVI does not introduce significant uncertainty for the critical heat flux assessment.

For the critical heat flux assessment, the peak core heat flux is applied to simulate the hot assembly condition in a conservative manner. No credit is taken for increased flow in the hot assembly, which is known to occur in rod bundles.

The correlation applied for this assessment is from vertical tube data (Chang, S. H., et al., "A study of critical heat flux for low flow of water in vertical round tubes under low pressure," Nuclear Engineering and Design, 132, 225-237, 1991). Chang measured critical heat flux in 0.72 m long vertical tubes at low pressure (atmospheric) and velocity conditions. The experiments considered tubes with diameters of 6 mm and 8.8 mm. The correlation was then validated against other low-pressure, low-flow critical heat flux data, which include points at higher pressure and wider range of mass flux. The applicable range of the correlation is shown below.

Correlation	Pressure Range	Mass Flux Range
Chang, et al. (1991)	0.1 to 1.1 MPa	-231 to +250 kg/m^2-s

The Chang correlation recognizes two regimes depending on the mass flux. The main difference between the two is the mass flux dependence. They are as follows:

$$q^*_{CL} = q^*_{CF} + 0.01351\left(D^*\right)^{-0.473}\left(L/D\right)^{-0.533}\left|G^*\right|^{1.45} \quad \text{for low } G^*$$

and,

$$q^*_{CH} = q^*_{CF} + 0.05664\left(D^*\right)^{-0.247}\left(L/D\right)^{-0.501}\left|G^*\right|^{0.77} \quad \text{for high } G^*$$

The first term of the above correlations is,

$$q^*_{CF} = 1.61\left(\frac{A}{Ah}\right)\frac{\left(D^*\right)^{0.5}}{\left[1+\left(\frac{\rho_g}{\rho_l}\right)^{0.25}\right]^2}$$

where A is the flow area and Ah is the heated area.

The dimensionless critical heat flux is calculated as,

$$q^*_{CHF} = \min(q^*_{CL}, q^*_{CH})$$

Dimensionless critical heat flux, G, and D are defined as,

$$q^*_{CHF} = \frac{q^*_{CHF}}{h_{fg}\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$G^* = \frac{G}{\sqrt{\lambda \rho_g g \Delta\rho}}$$

$$D^* = \frac{D}{\lambda}$$

where λ is the length scale of the Taylor instability:

$$\lambda = \sqrt{\frac{\sigma}{g\Delta\rho}}$$

Conservative application of this correlation with the AP1000 parameters indicates that the peak AP1000 heat flux during this period is at least 40 percent below the predicted critical heat flux.

This critical heat flux assessment addresses core cooling during a time period where the NOTRUMP computer code may not conservatively predict the core average void fraction. The requirements of 10 CFR 50.46 are met during this period since this critical heat flux assessment indicates peak core heat flux is less than critical heat flux. Cladding temperatures will remain near the coolant saturation temperature, well below the 10 CFR 50.46 peak cladding temperature limit.

3.8 REFERENCES

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2. WCAP-10054-P-A, "Westinghouse SB ECCS Evaluation Model Using the NOTRUMP Code," Lee, N., et al., 1985.

3. WCAP-14807, Revision 5, "NOTRUMP Final Verification and Validation Report for AP600," Fittante, R. L., et al., 1998.

4. WCAP-14206 "Applicability of NOTRUMP Computer Code to AP600 SSAR Small Break LOCA Analysis," Kemper, R. M., 1994.

5. NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," U.S. Nuclear Regulatory Commission, September 1998.

6. NSD-NRC-99-5839, "1998 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10CFR50.46 (a)(3)(ii)," Letter from J. S. Galembush (Westinghouse) to J. S. Wermiel (NRC), July 15, 1999.

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12. OSU-APEX-03002, Revision 0, OSU Facility Description Report for AP1000 Simulation Series, K. C. Abel, et al., Oregon State University, Department of Nuclear Engineering, May 12, 2003.

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15. Lombardi, C., Maran, L., Oriani, L., Ricotti, M., "CESNEF-3 Pressure Drop Correlation for Gas-Liquid Mixture Flowing Upflow in Vertical Ducts," ATTI XVIII Congresso Nazionale sulla Trasmissione del Calore (18th UIT National Heat Transfer Conference): Cernobbio, June 28–30, 2000.

16. Crane Technical Paper 410, Flow of Fluids Through Valves, Fittings, and Pipe, 1976.

Table 3-1 NOTRUMP Issue Assessment Summary for AP1000

Component Phenomenon	Assessment Results	AP600 Treatment	Comments	AP1000 Treatment
ADS-4:				
Two-phase pressure drop	Minimal; due to lack of momentum flux terms, underpredicted pressure drop.	Apply IRWST level penalty.[1] Upper bound loss coefficients.	Flow out ADS-4 is overpredicted, resulting in early Pressurizer drain and IRWST initiation.	Upper bound loss coefficients. Use ADS-4 resistance increase developed via detailed stand-alone momentum flux model of the ADS-4 flow path.
COLD LEGS:				
Phase separation at tees	Minimal, but conservative.	No change.	Balance line refilling delays CMT drain and subsequent ADS actuation.	No change.
CMT:				
Thermal stratification	Minimal, but conservative.	No change.	Inability to accurately track thermal stratification increases CMT exit temperature, reduces core subcooling.	No change.
DOWNCOMER:				
Level	Minimal for DEDVI.	Apply IRWST level penalty. Range Cd for break to assure limiting case found.	Downcomer model does not predict 2-dimensional temperatures. Excess condensation during IRWST injection.	Apply ADS-4 resistance increase. Range Cd for break to assure limiting case found. Downcomer misprediction does not impact core level response.

Note:

1. Level penalty is indirect correction for most significant deficiency, lack of momentum flux in ADS-4. All SAR cases run with increase ADS-4 resistance to confirm level penalty approach.

Table 3-1 NOTRUMP Issue Assessment Summary for AP1000 (cont.)

Component Phenomenon	Assessment Results	AP600 Treatment	Comments	AP1000 Treatment
HOT LEGS:				
Stratification, phase separation at tees	Minimal due to ad hoc model; impact is small.	Apply IRWST level penalty.	Liquid flow out ADS-4 is controlled by constant system inventory, inlet flow, self-correcting system.	Apply ADS-4 resistance increase. Hot leg/upper plenum sensitivity study.
PRESSURIZER AND SURGE LINE:				
CCFL	Minimal but conservative provided vapor flow is correct.	No change; given correct or high vapor flow, CCFL is conservative.	Rapid draining through surge line caused by low vapor flow due to low pressure drop through ADS-4.	ADS-4 resistance increase application.
Level swell	Minimal non-conservative during draining.	Apply IRWST level penalty.	Rapid draining due to poor ADS-4 pressure drop prediction; confirmed by studies with increased ADS-4 resistance.	ADS-4 resistance increase application. Additional validation of NOTRUMP level swell model against full-scale test data.
STEAM GENERATOR:				
Heat transfer	Minimal, but conservative	No Change.	Underprediction in PRHR, CMT increases SG heat transfer/reliance on ADS.	No Change.

Table 3-1 NOTRUMP Issue Assessment Summary for AP1000 (cont.)

Component Phenomenon	Assessment Results	AP600 Treatment	Comments	AP1000 Treatment
PRHR:				
Heat transfer	Minimal, conservative if primary flow is low.	Remove PRHR after ADS-3 and check PRHR flow.	Heat transfer not overpredicted as long as primary side is limiting.	Remove PRHR after ADS-3 and check PRHR flow.
Recirculation flow	Minimal, conservative if primary flow is low.	Remove PRHR after ADS-3 and check PRHR flow.	Underpredicted flow reduces PRHR heat transfer.	Remove PRHR after ADS-3 and check PRHR flow.
NON-CONDENSABLE GAS INJECTION:				
Accumulator nitrogen injection	Model not available in code.	PRHR removed prior to the introduction of non-condensable gases.	Removal of PRHR conservatively bounds the effect of the introduction of non-condensable gases.	PRHR removed prior to the introduction of non-condensable gases.

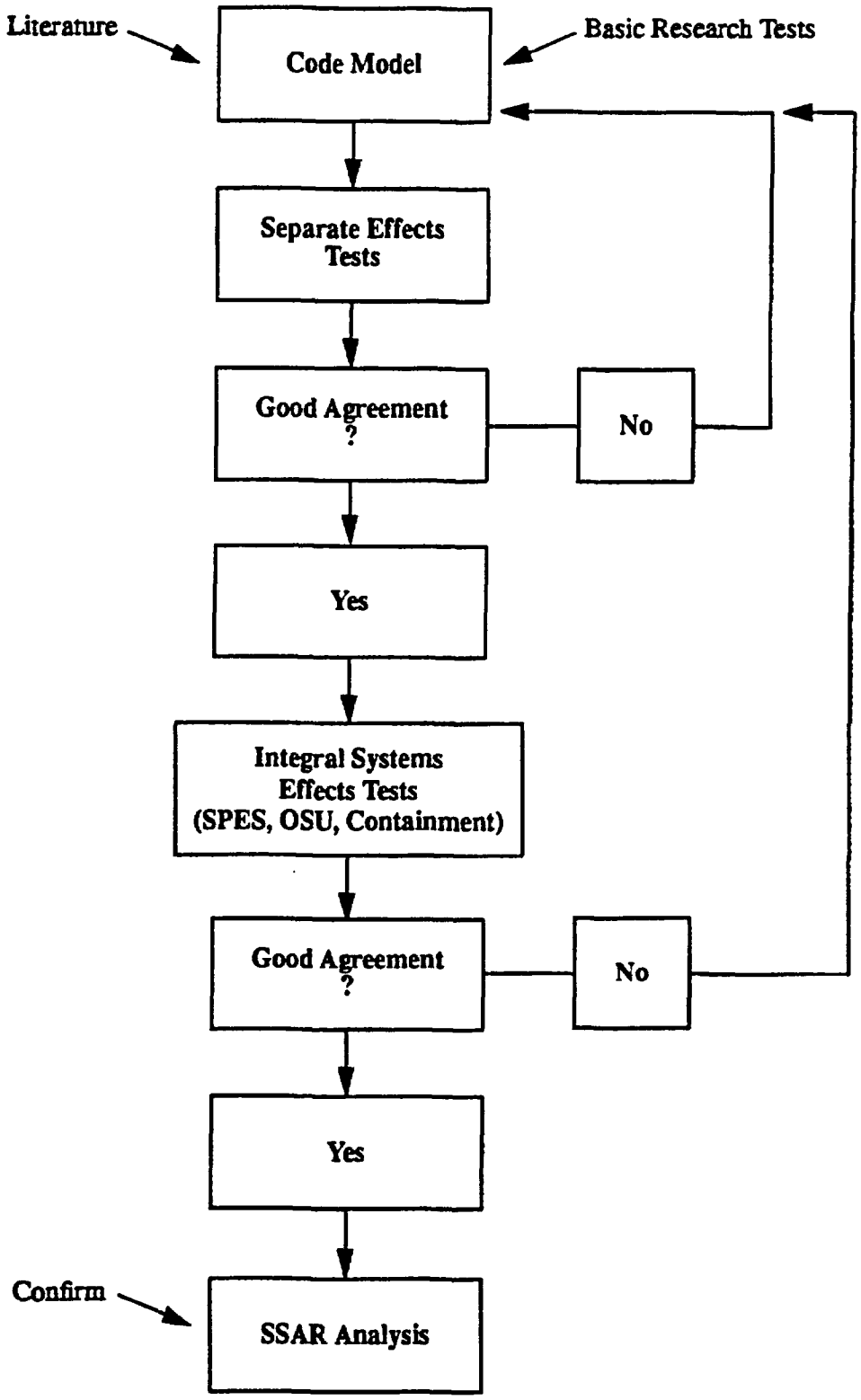

Figure 3-1 Model Development and Verification Process for Code Validation

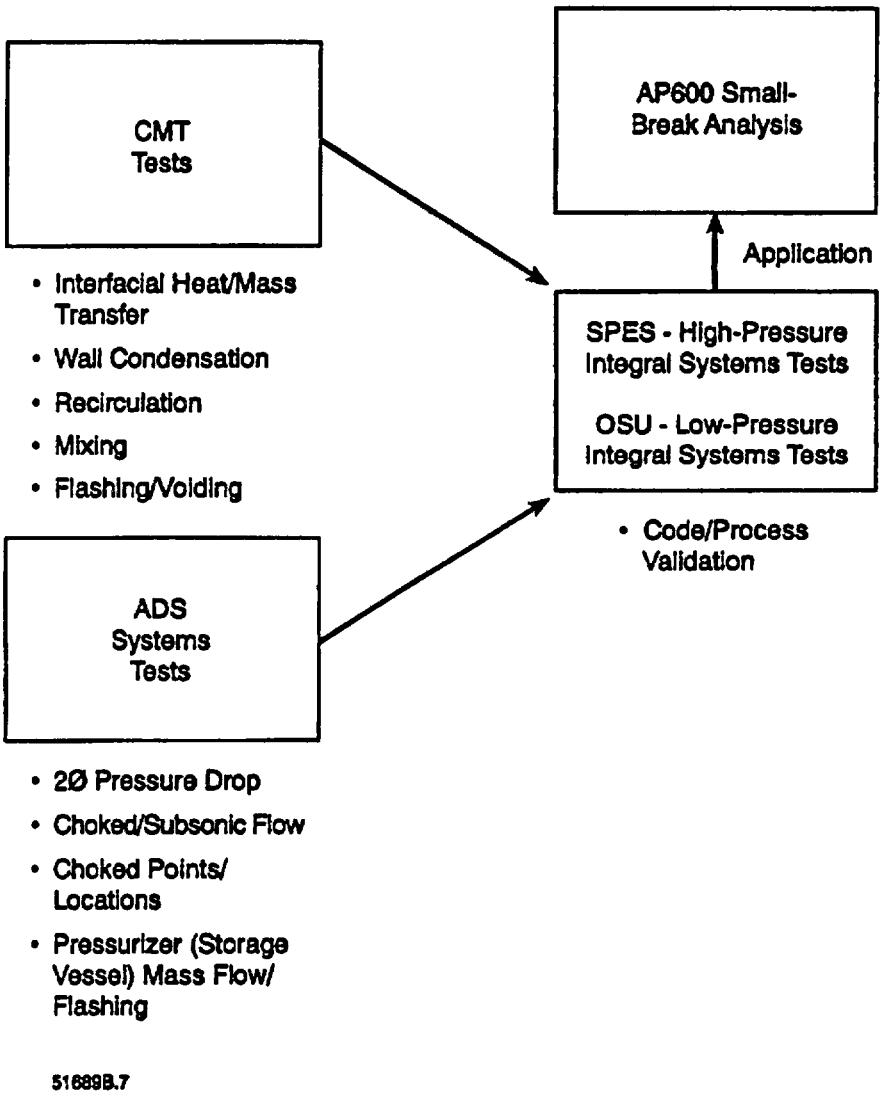

Figure 3-2 NOTRUMP Verification with Separate Effects Tests and Validation with Integral Systems Tests

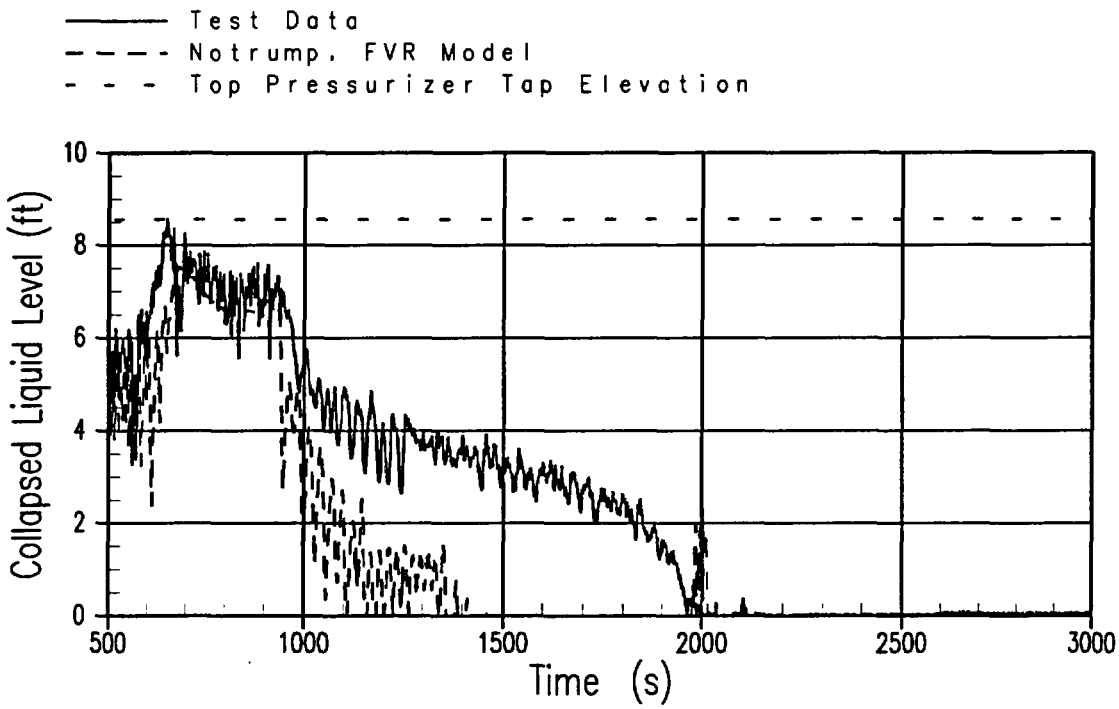

Figure 3-3 OSU Test SB18 2-Inch Cold Leg Break Pressurizer Level (Relative to Bottom Tap)

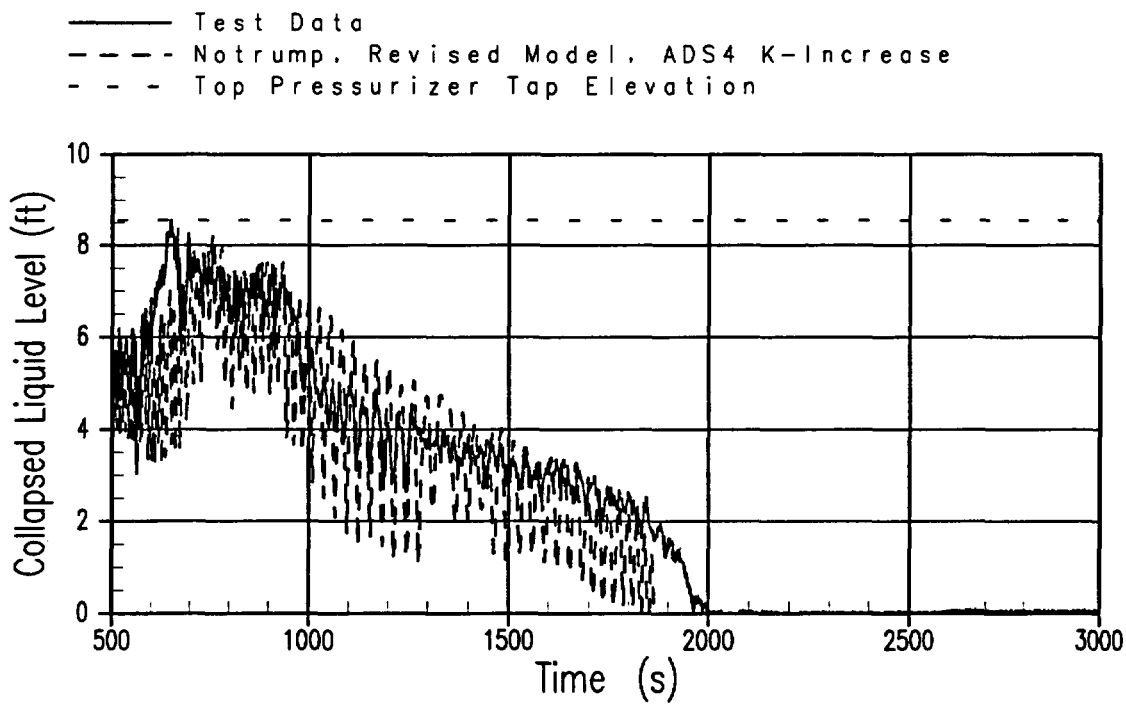

Figure 3-4 OSU Test SB18 2-Inch Cold Leg Break Pressurizer Level (Relative to Bottom Tap)

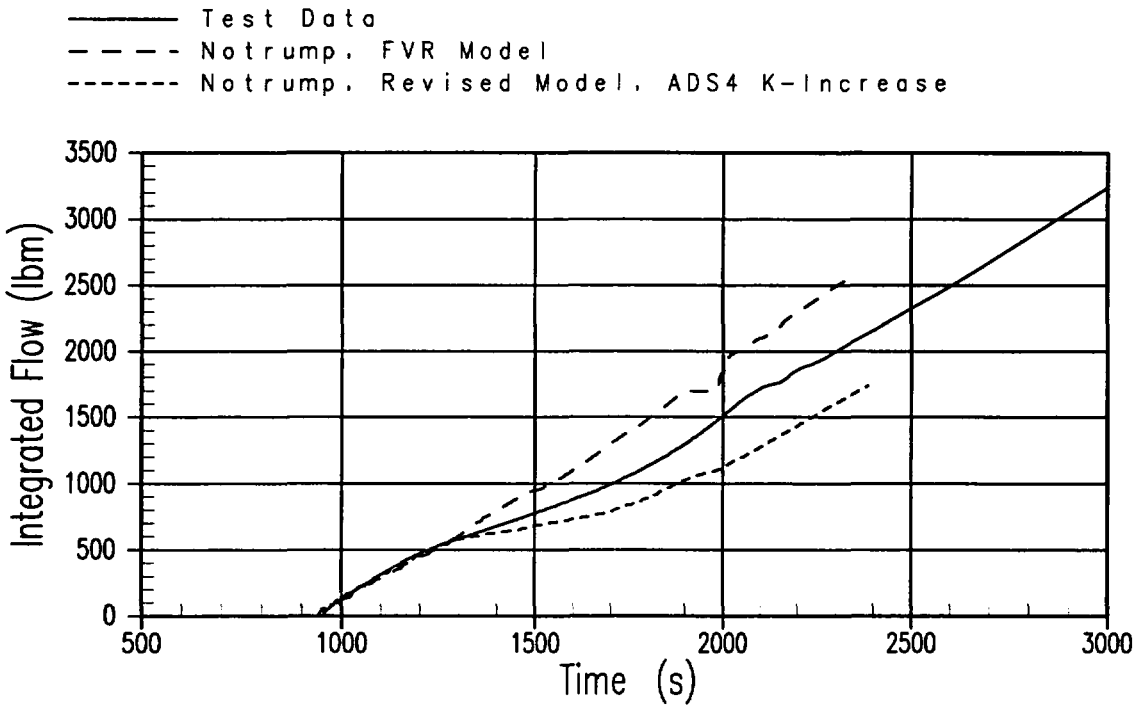

Figure 3-5 OSU Test SB18 2-Inch Cold Leg Break ADS Stage 4 Integrated Flows

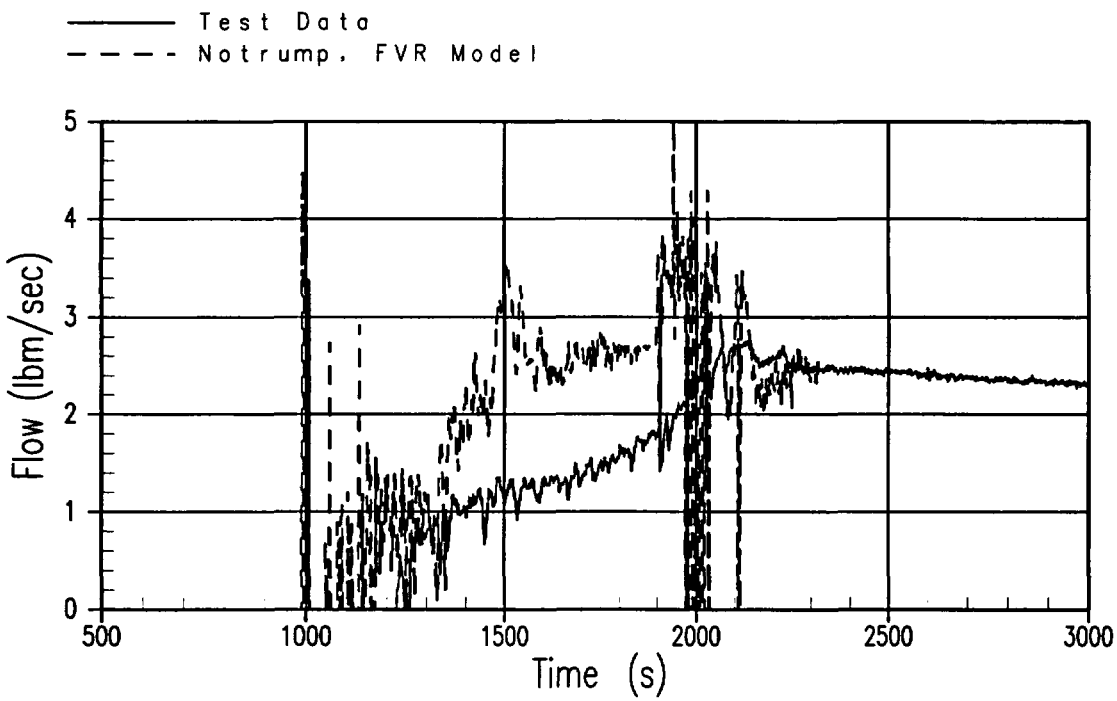

Figure 3-6 OSU Test SB18 2-Inch Cold Leg Break IRWST Injection Line Mass Flow

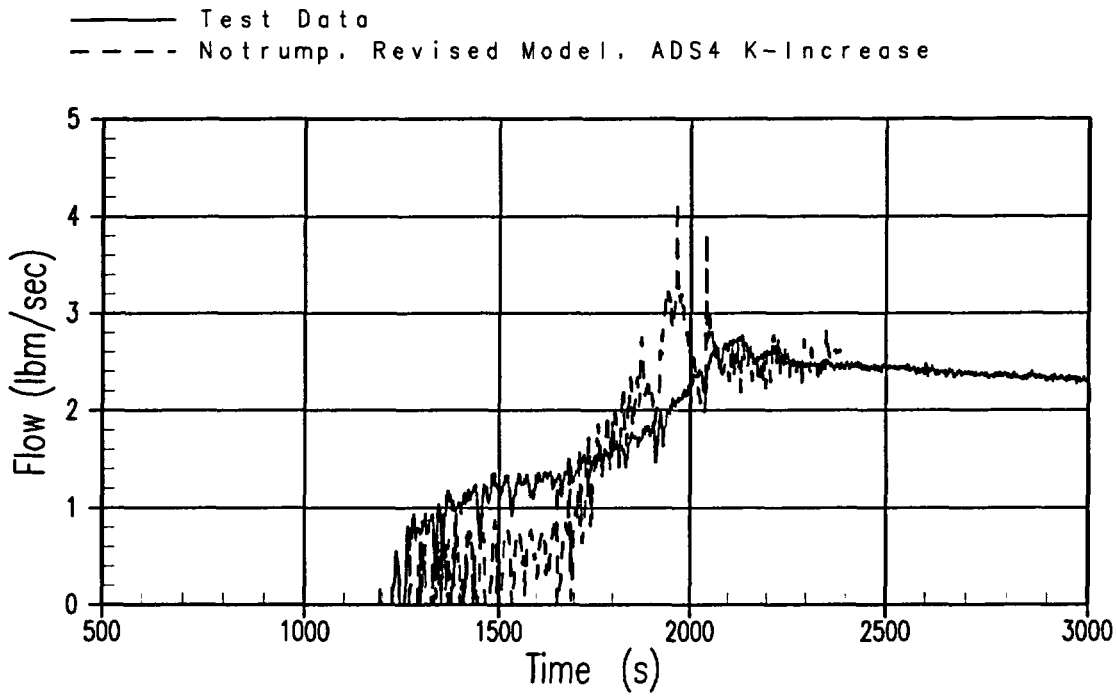

Figure 3-7 OSU Test SB18 2-Inch Cold Leg Break IRWST Injection Line Mass Flow

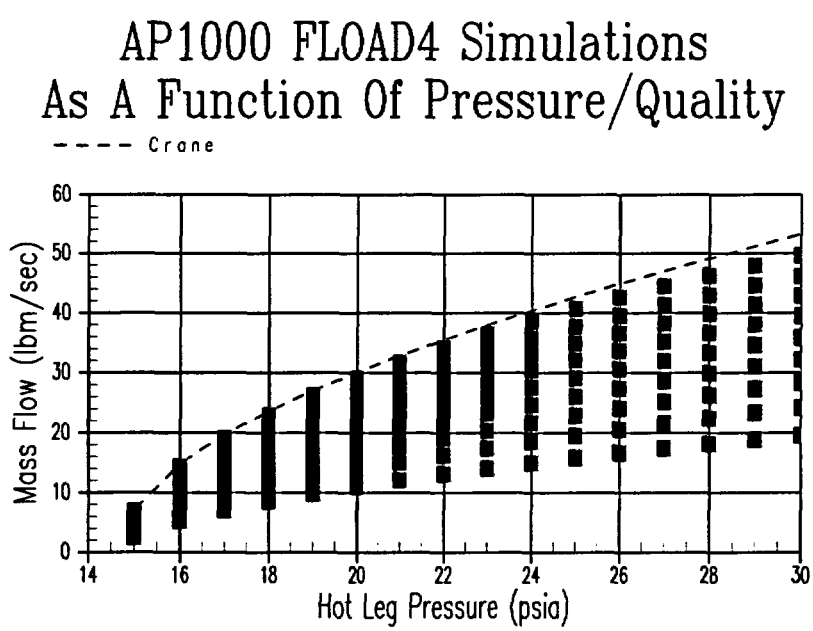

Figure 3-8 FLOAD4 Results and Comparison to Crane

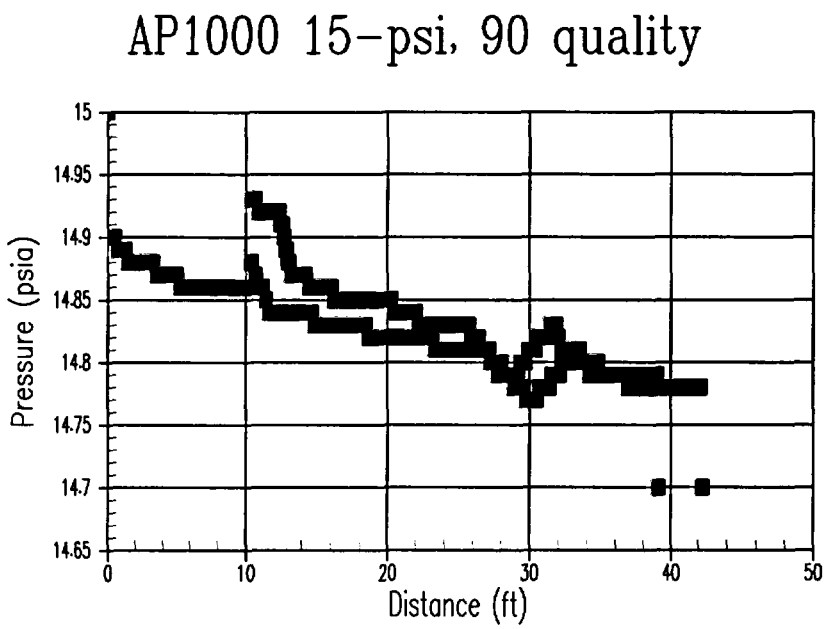

Figure 3-9 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 15 psia, 90-Percent Quality

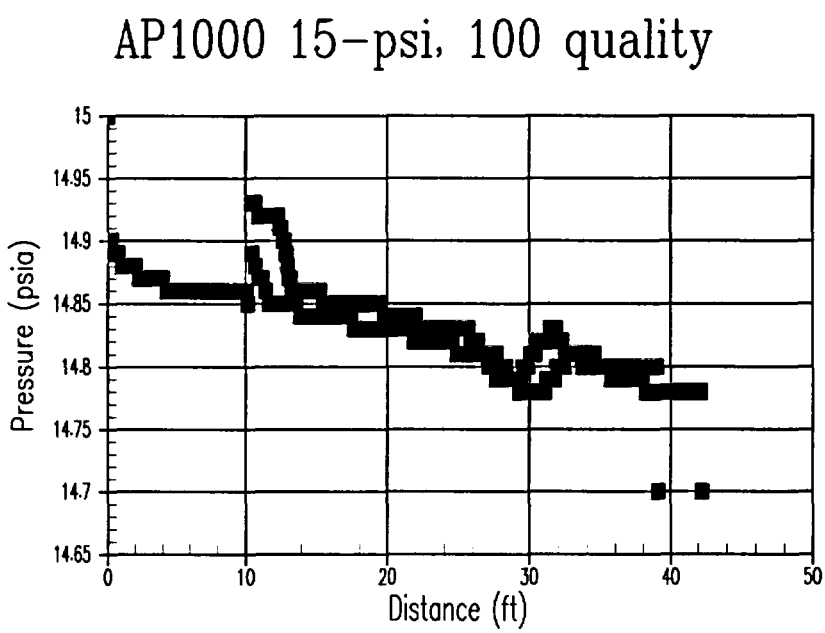

Figure 3-10 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 15 psia, 100-Percent Quality

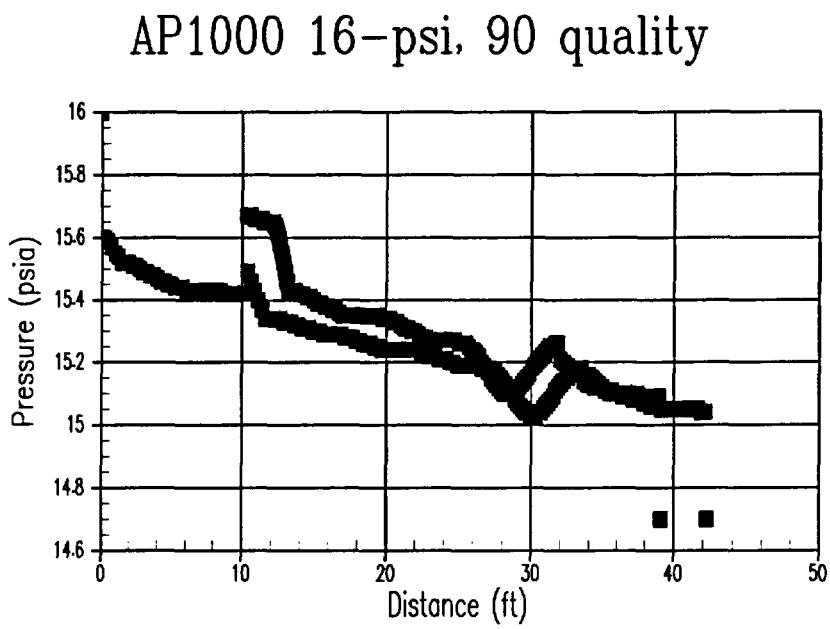

Figure 3-11 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 16 psia, 90-Percent Quality

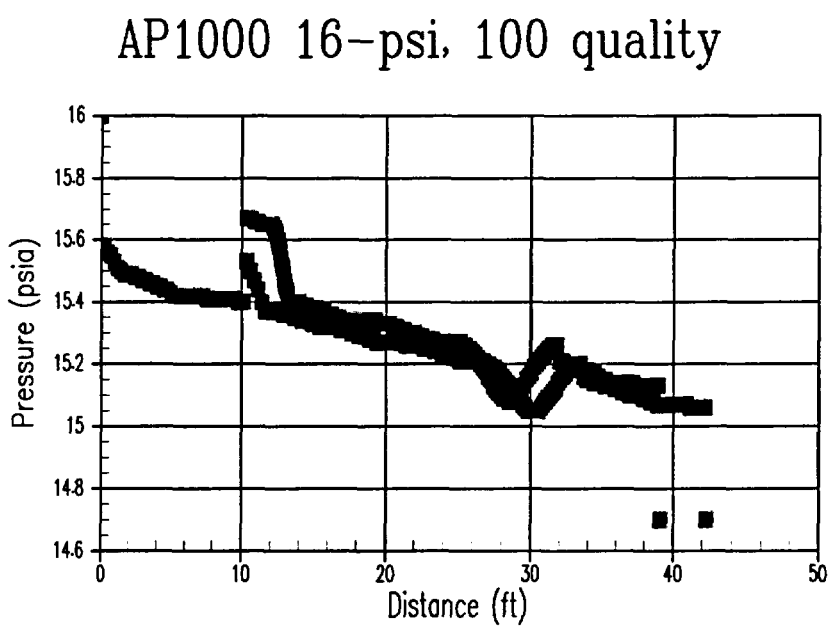

Figure 3-12 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 16 psia, 100-Percent Quality

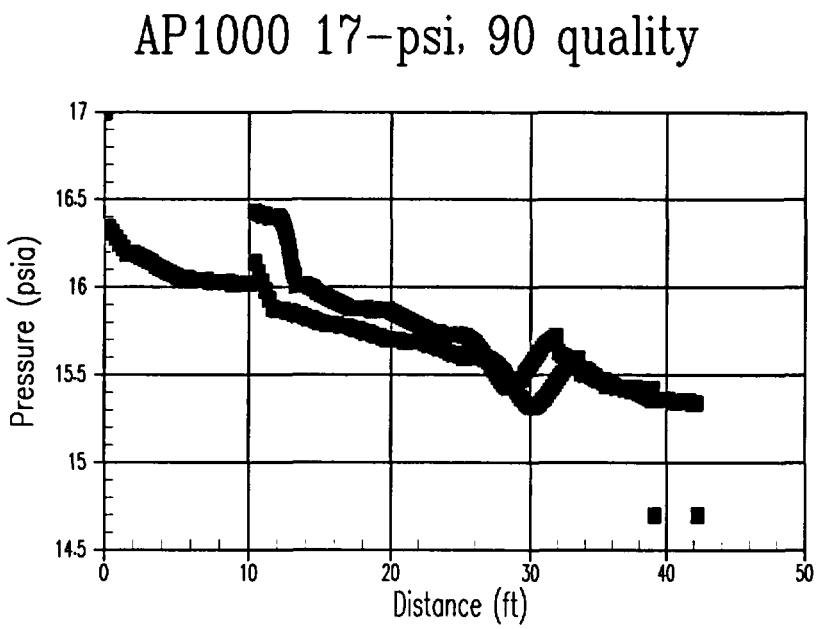

Figure 3-13 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 17 psia, 90-Percent Quality

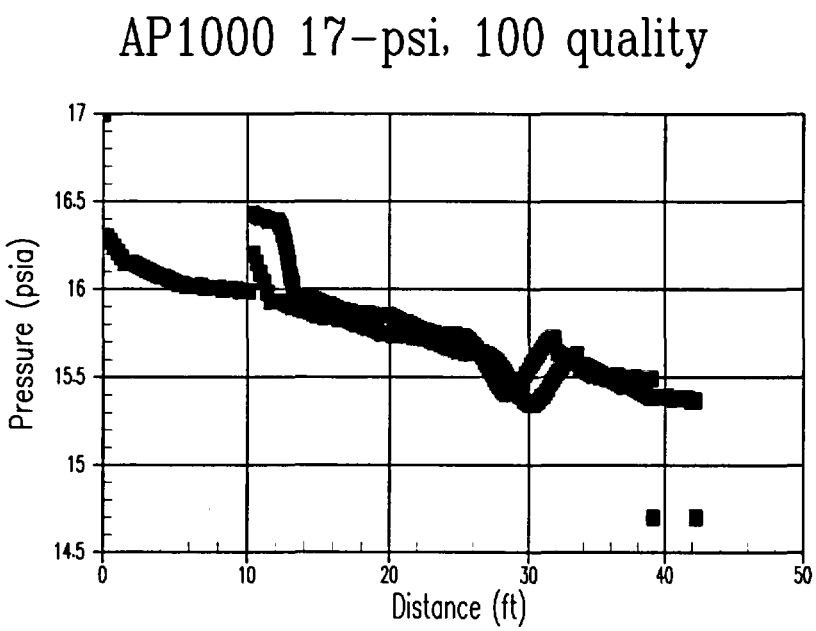

Figure 3-14 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 17 psia, 100-Percent Quality

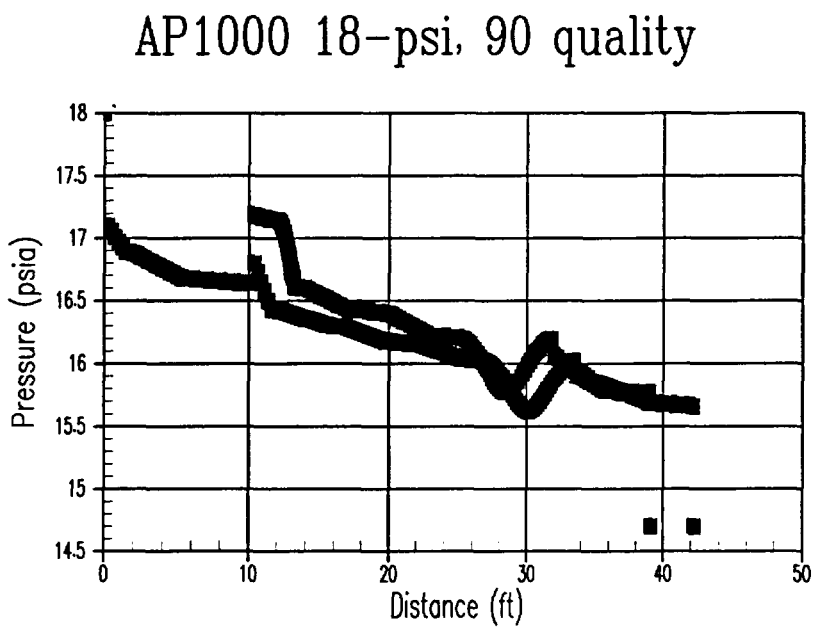

Figure 3-15 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 18 psia, 90-Percent Quality

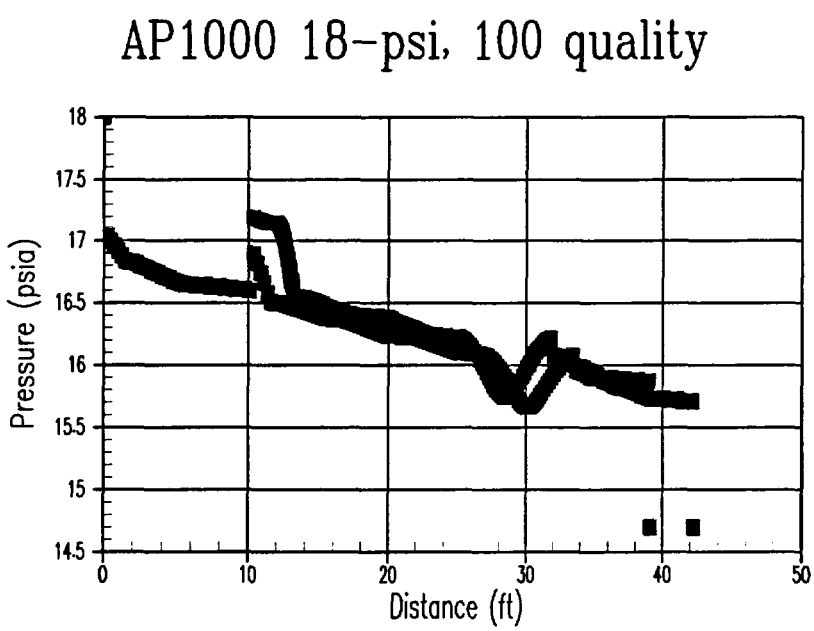

Figure 3-16 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 18 psia, 100-Percent Quality

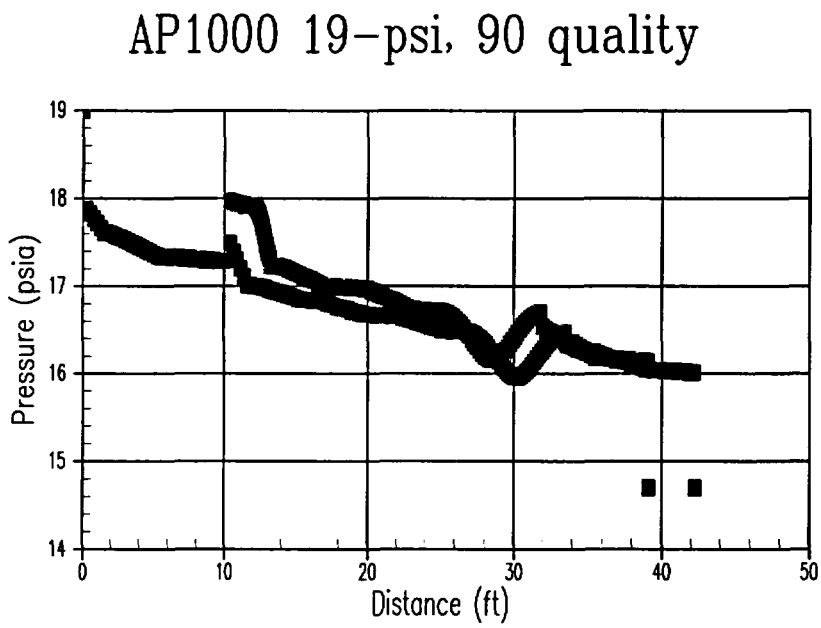

Figure 3-17 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 19 psia, 90-Percent Quality

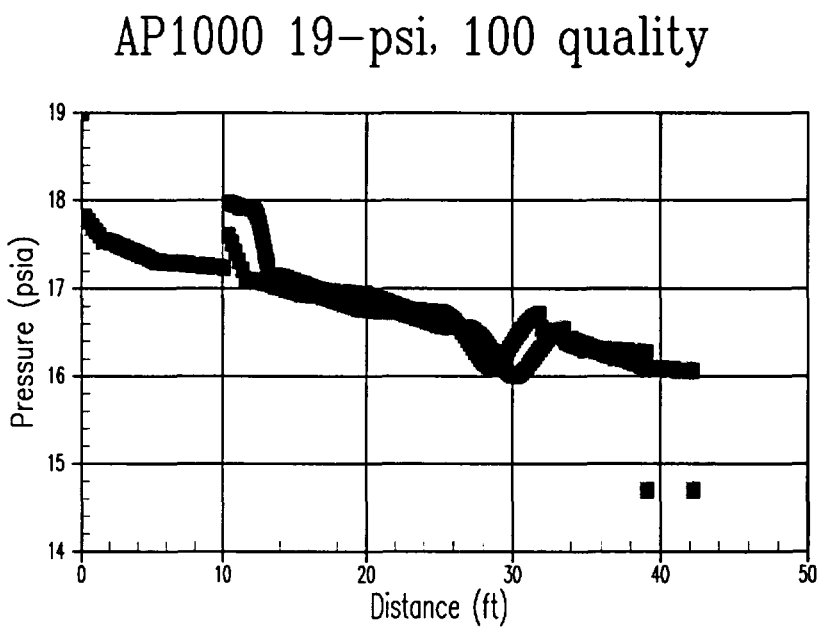

Figure 3-18 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 19 psia, 100-Percent Quality

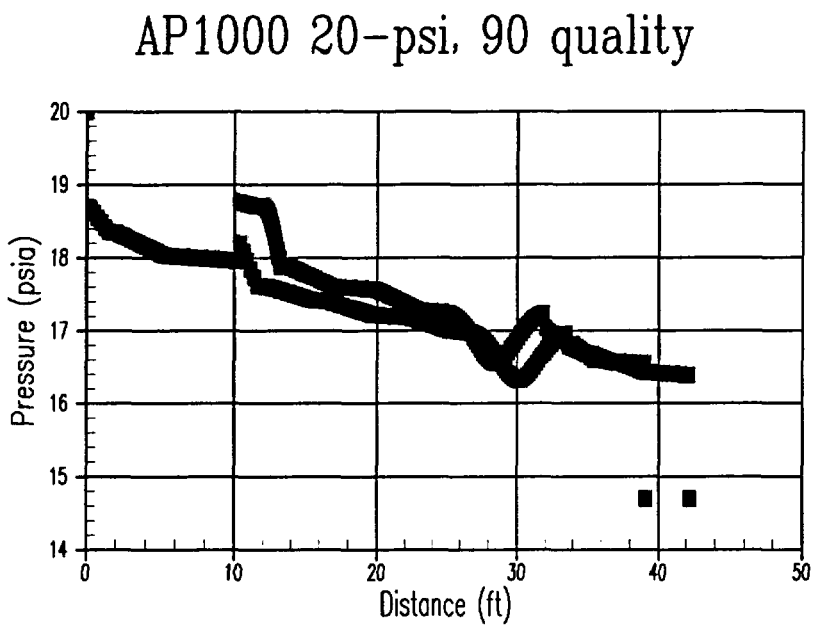

Figure 3-19 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 20 psia, 90-Percent Quality

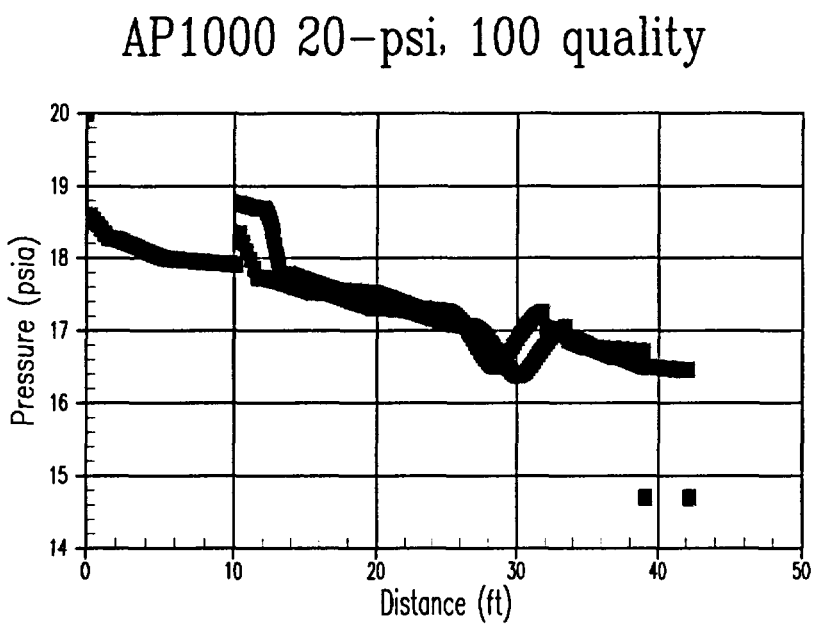

Figure 3-20 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 20 psia, 100-Percent Quality

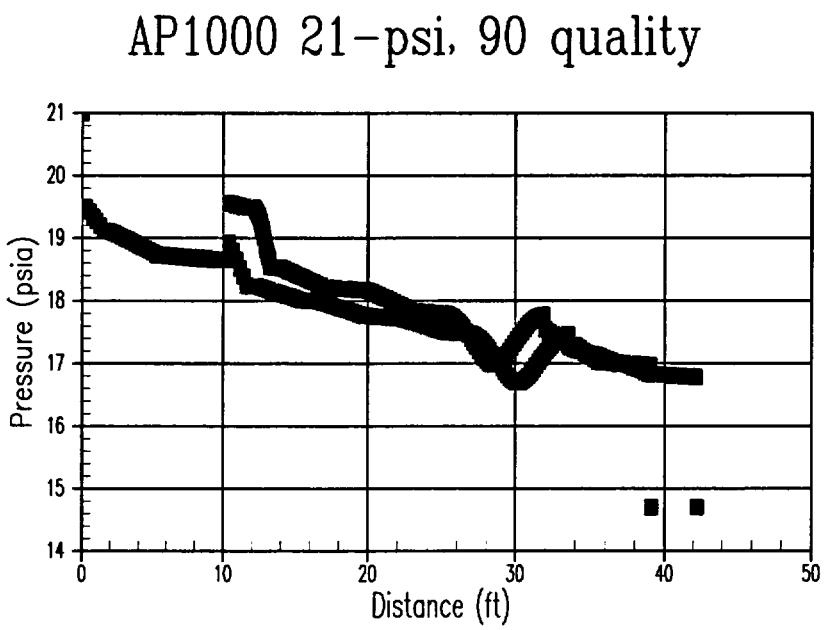

Figure 3-21 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 21 psia, 90-Percent Quality

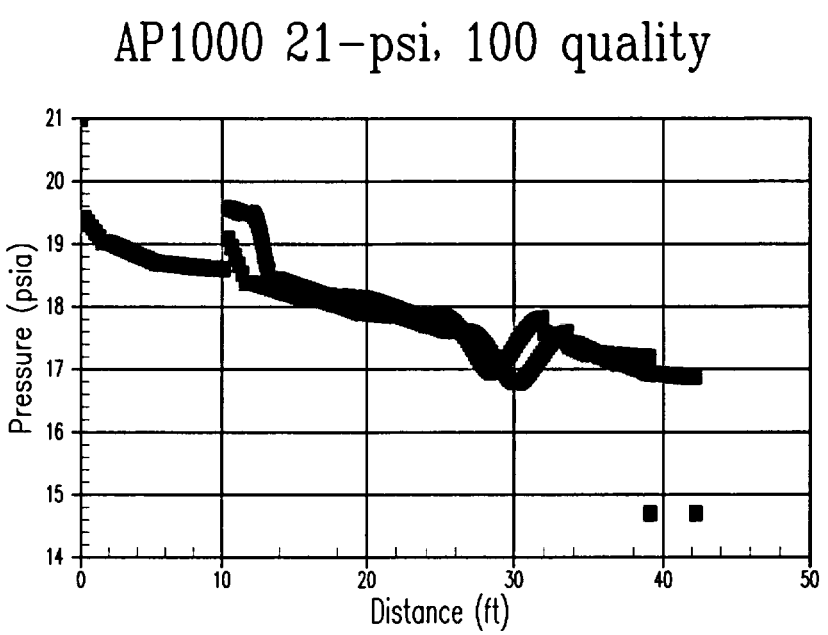

Figure 3-22 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 21 psia, 100-Percent Quality

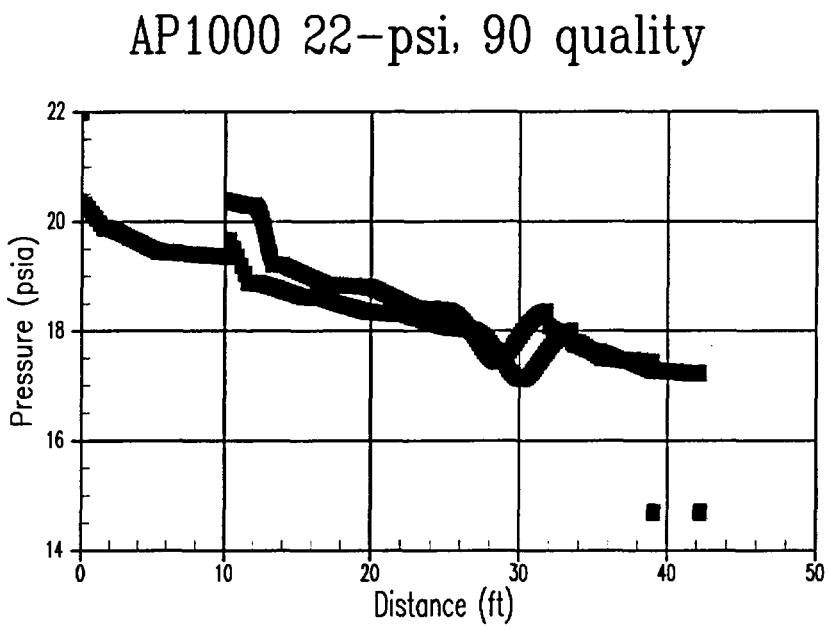

Figure 3-23 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 22 psia, 90-Percent Quality

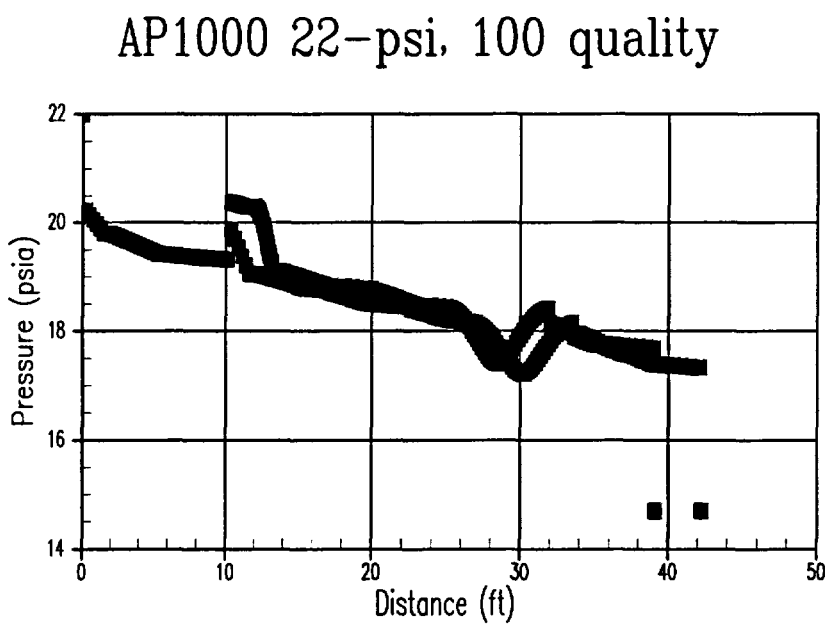

Figure 3-24 FLOAD4 ADS-4 Pressure Versus Distance from Hot Leg at 22 psia, 100-Percent Quality

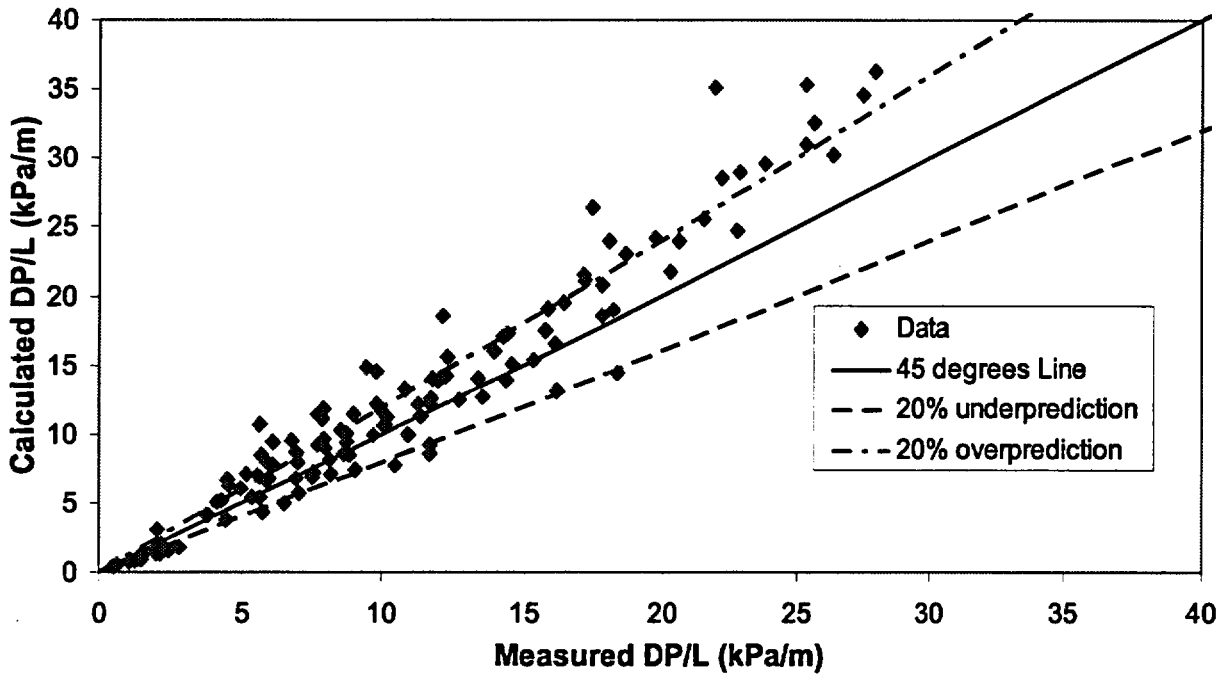

Figure 3-25 CESNEF-2 Versus Low-Pressure Data

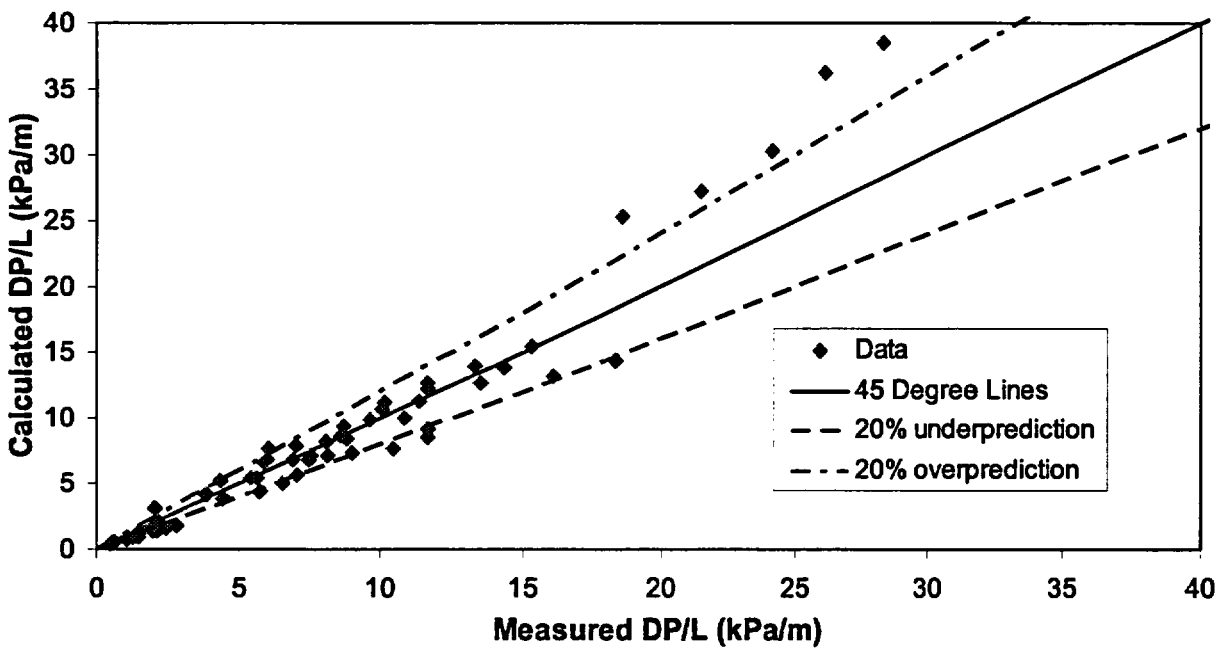

Figure 3-26 CESNEF Versus High-Quality Data (X > 70 Percent)

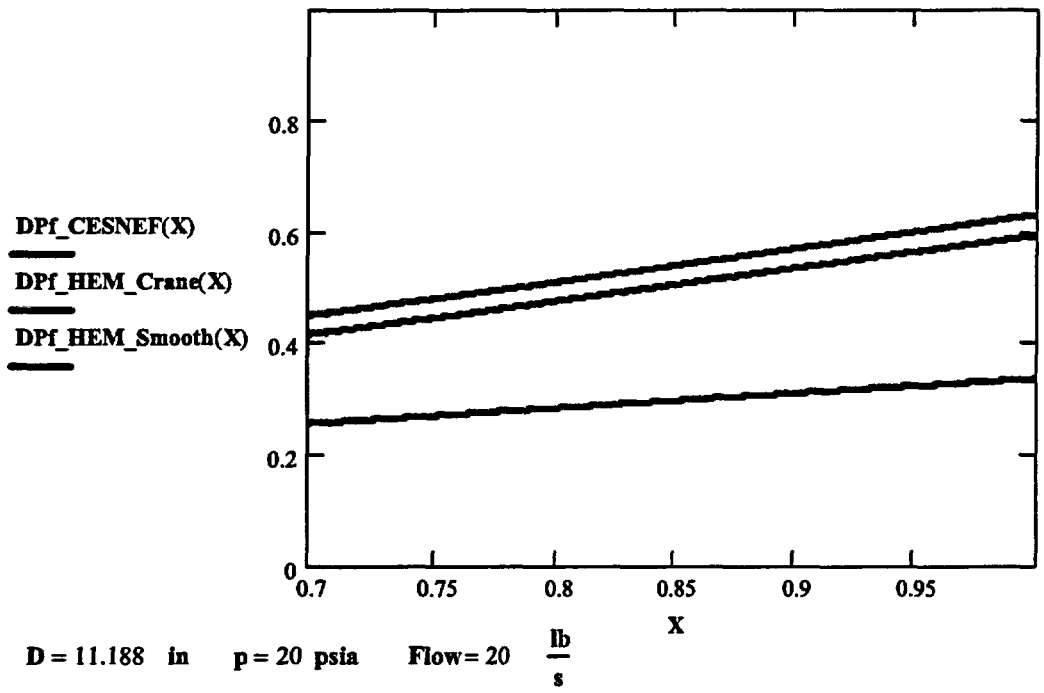

Figure 3-27 CESNEF-2 Comparison to Homogeneous Models

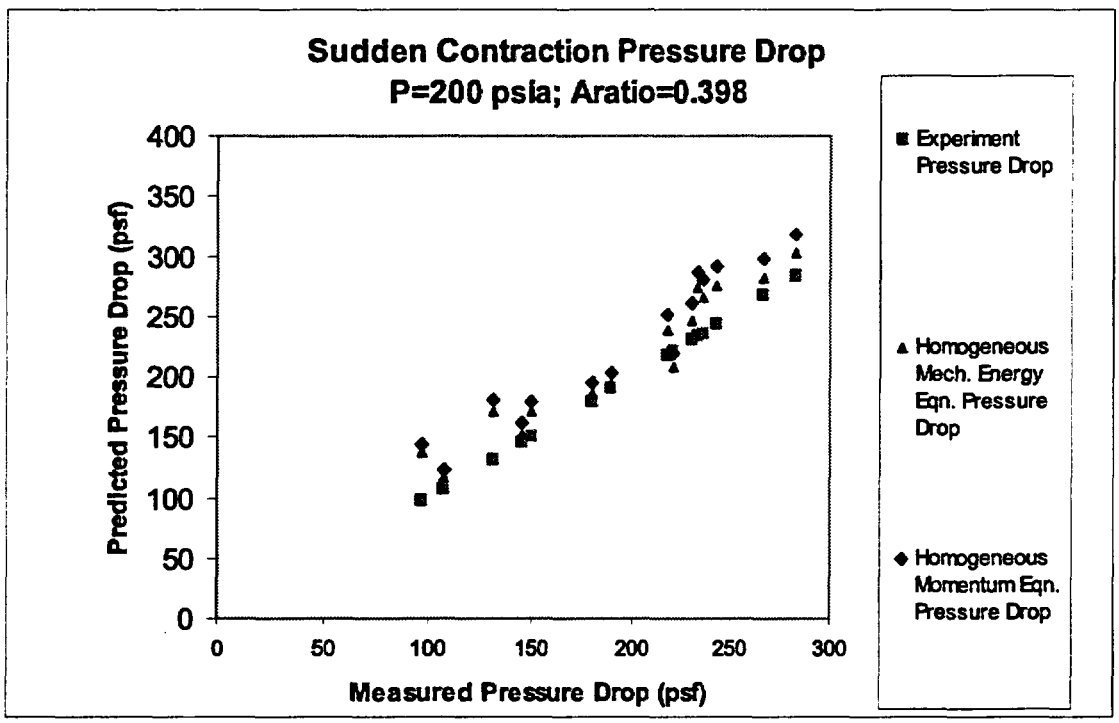

Figure 3-28 Steam-Water Data, Quality Range X = 0.0117-0.211

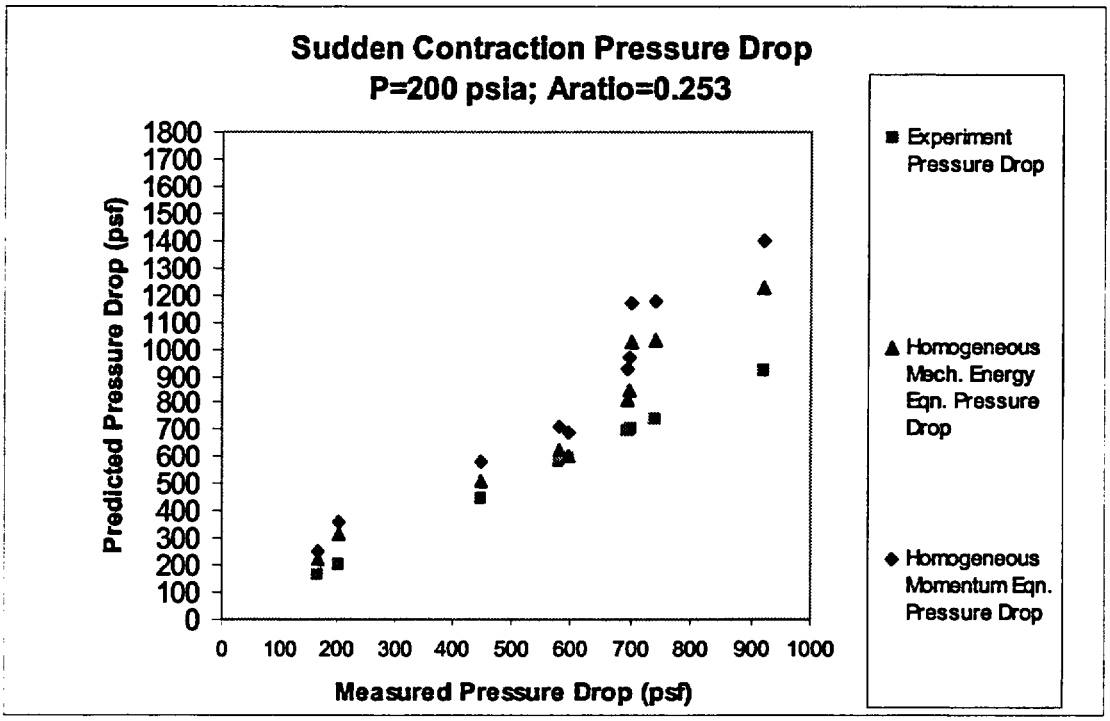

Figure 3-29 Steam-Water Data, Quality Range X = 0.0001-0.193

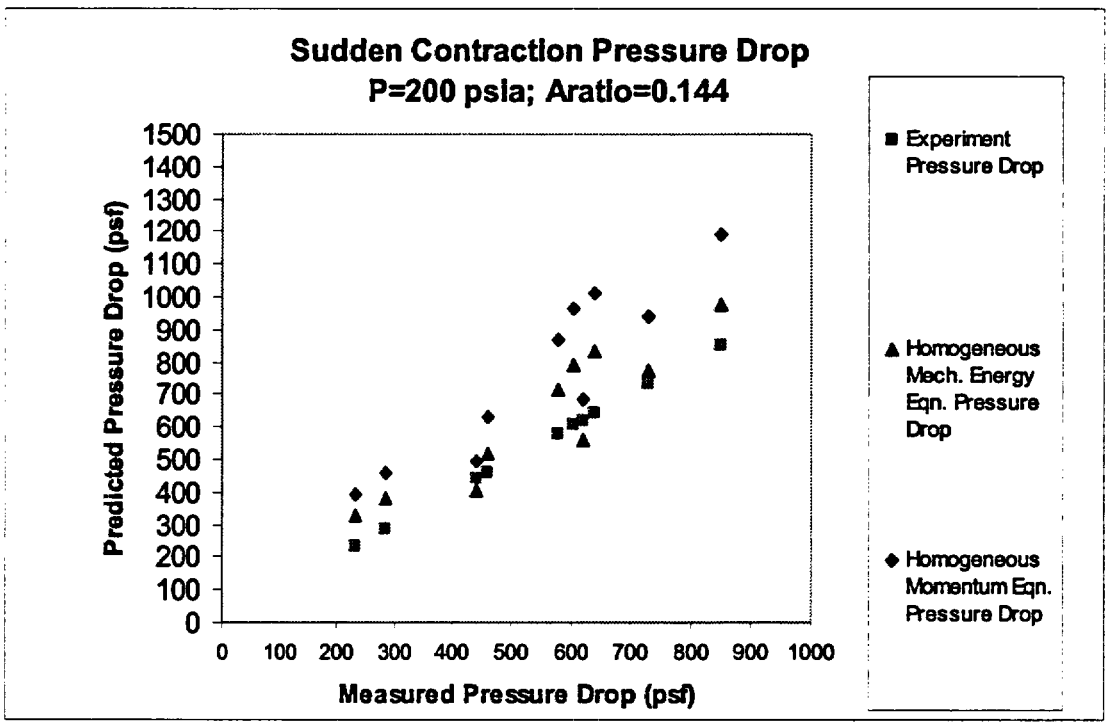

Figure 3-30 Steam-Water Data, Quality Range X = 0.0001-0.0858

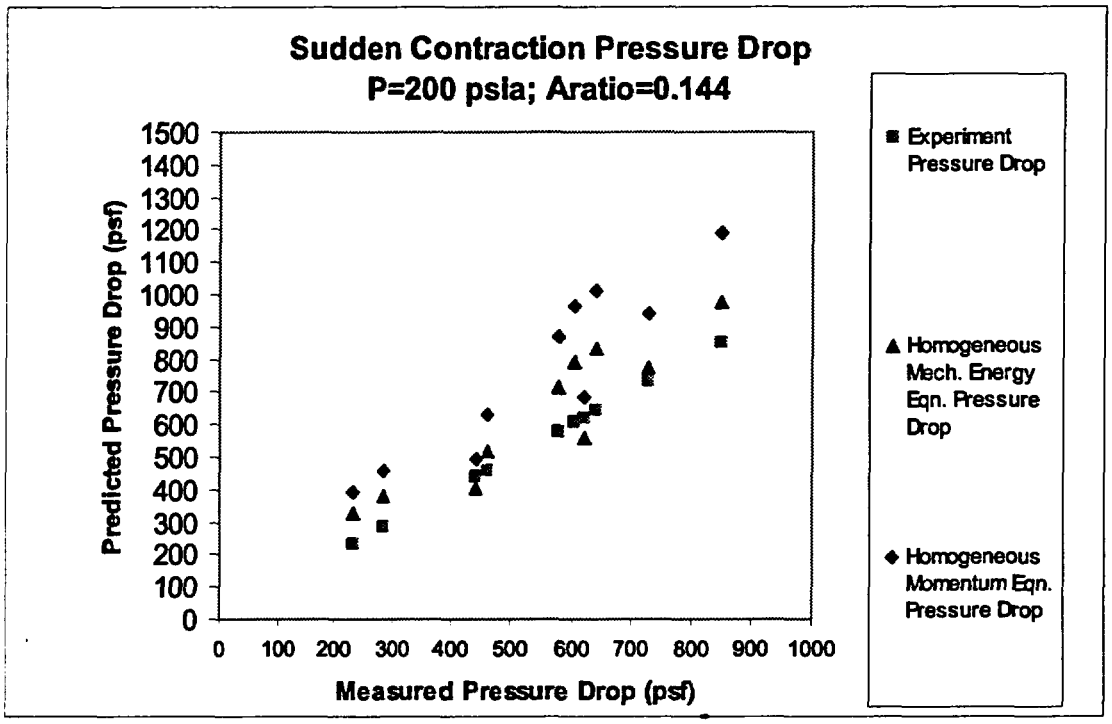

Figure 3-31 Air-Water Data, Quality Range X = 0.1-1.0

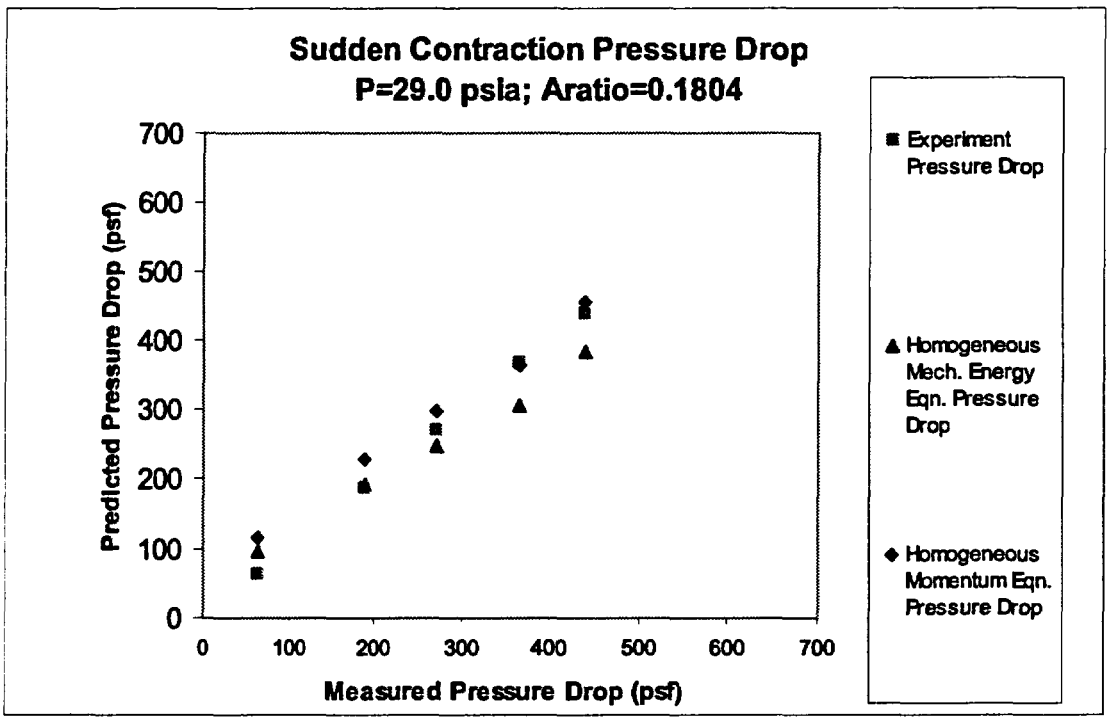

Figure 3-32 Air-Water Data, Quality Range X = 0.25-1.0

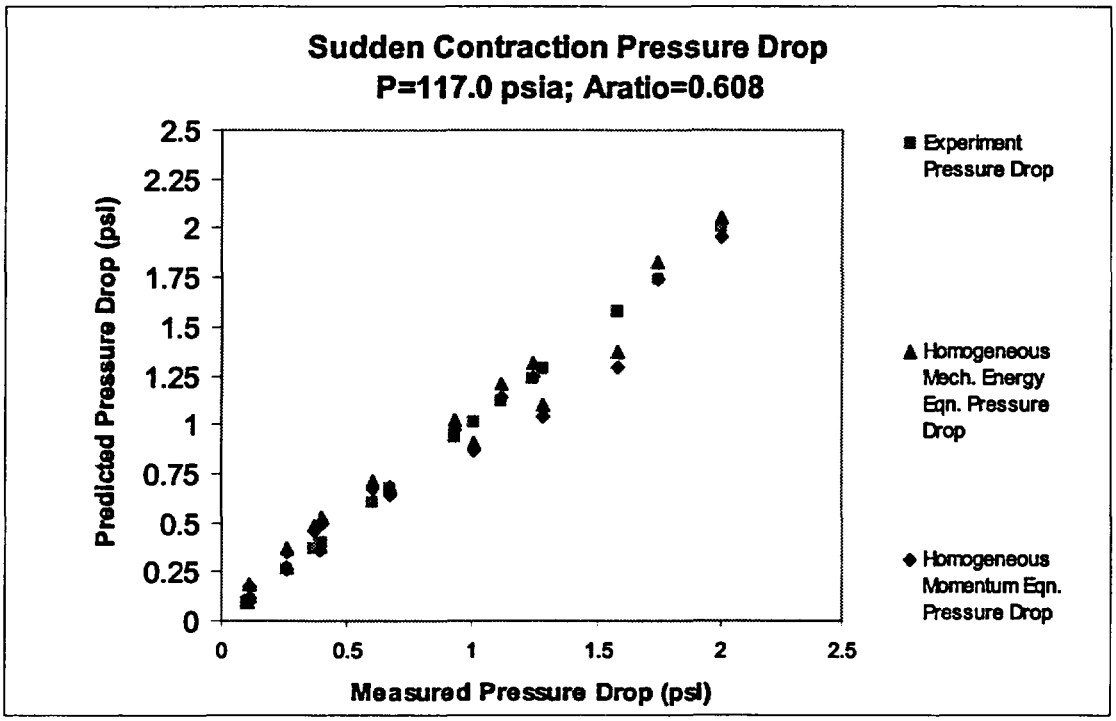

Figure 3-33 Ferrell and McGee Data

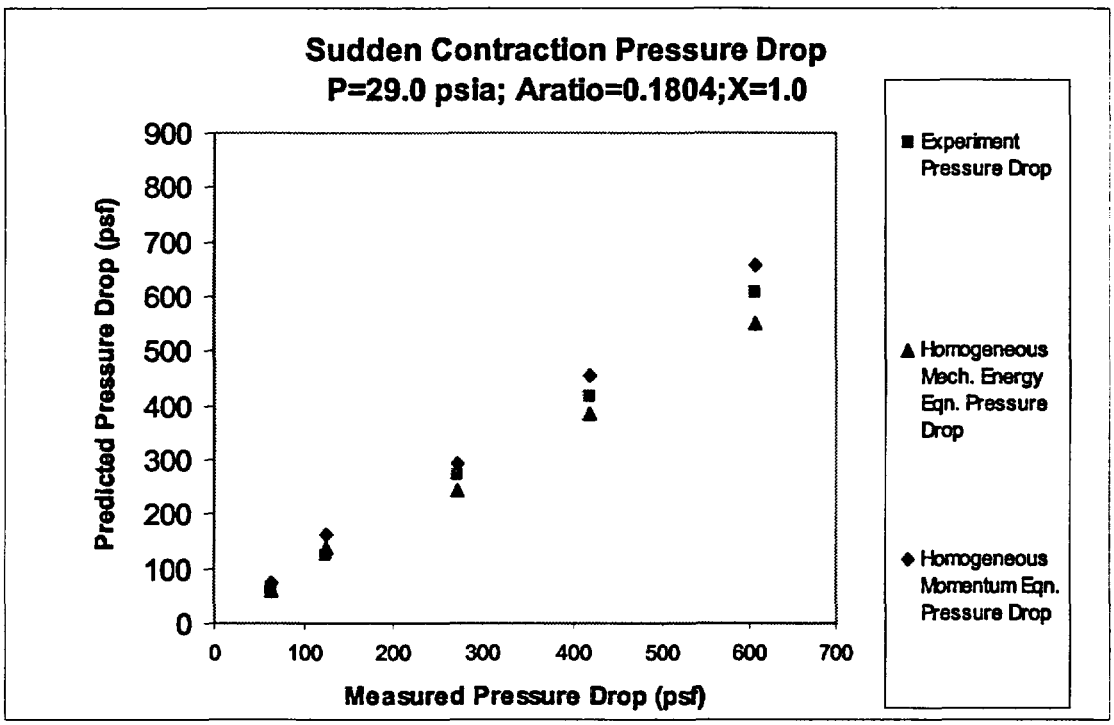

Figure 3-34 Air Data

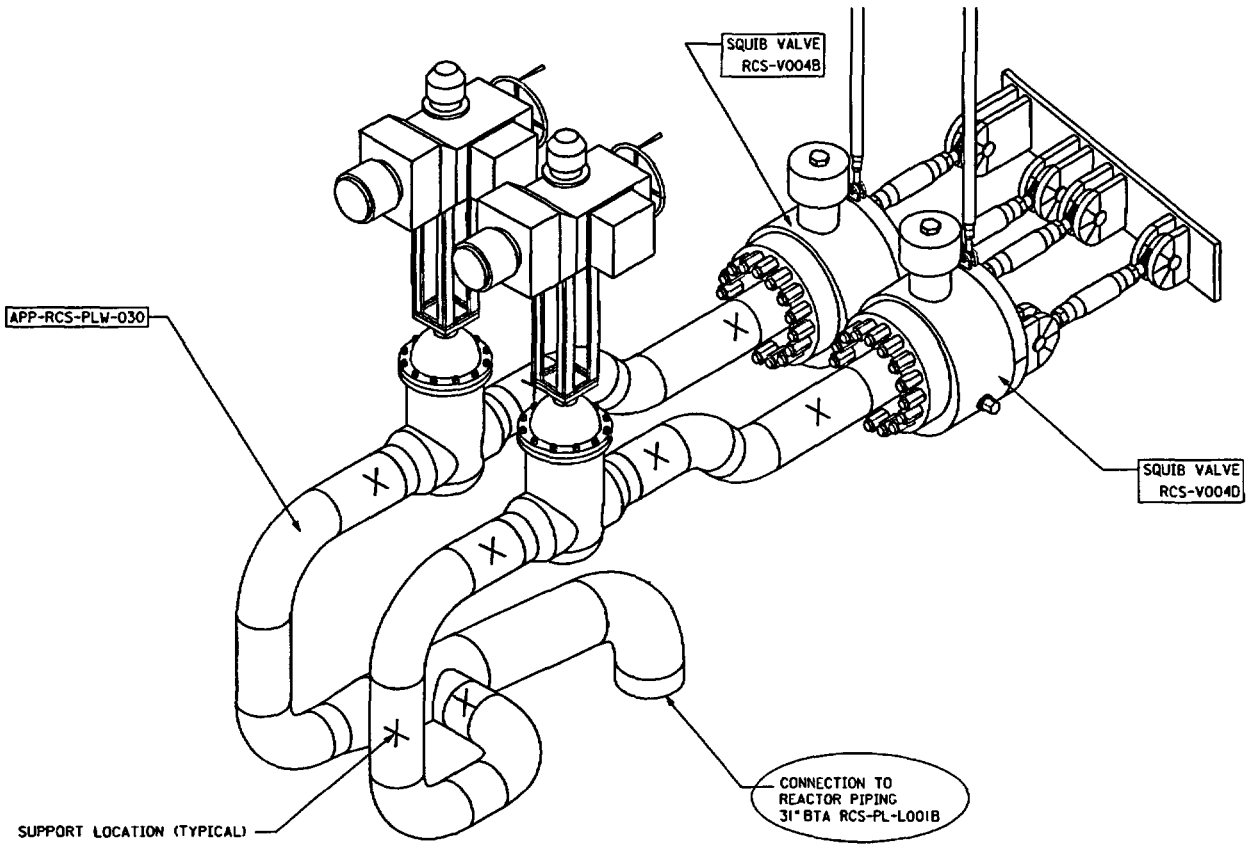

Figure 3-35 ADS-4 Piping Configuration

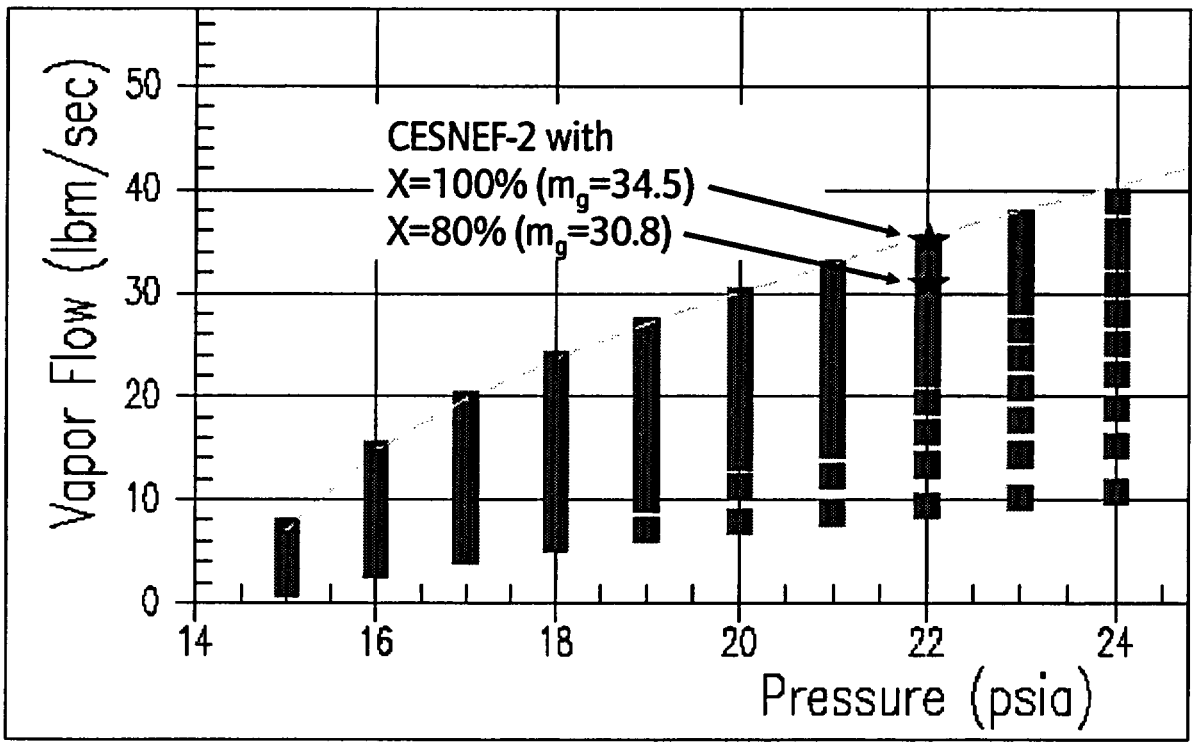

Figure 3-36 Results and Comparison with CESNEF-2

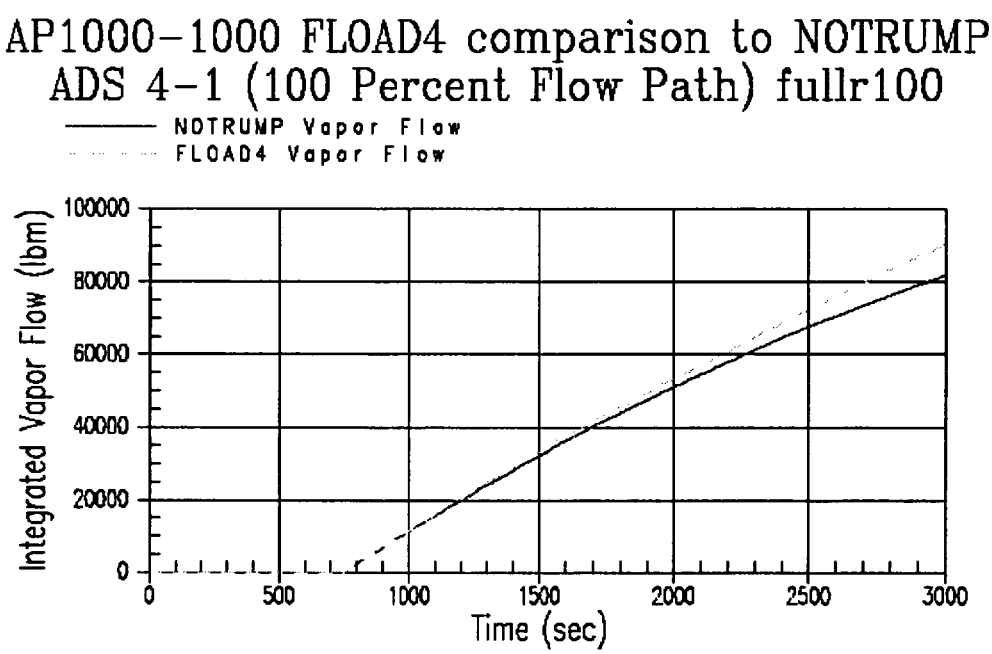

Figure 3-37 Comparison Between Integrated Vapor Flow Predicted by NOTRUMP and FLOAD4

4.0 LOFTRAN-AP CODE VALIDATION

4.1 INTRODUCTION

The original LOFTRAN computer code (Reference 1) was developed to simulate behavior in a multi-loop pressurizer water reactor with active safety systems during non-LOCA events. The code simulates a multi-loop system by modeling the reactor core and vessel, hot and cold leg, steam generator (tube and shell sides), pressurizer, and reactor coolant pumps, with up to four reactor coolant loops. The code has an extensive history of use in performing design and licensing basis non-LOCA analyses and has been reviewed and approved for use in non-LOCA analyses by the U.S. NRC. The code is currently used for licensing analyses in support of operating plant fuel reloads and plant upgrades (upratings, steam generator replacement programs).

Several specialized versions of LOFTRAN have been developed for steam generator tube rupture (SGTR) analyses and for non-LOCA analyses that use passive safety systems for event mitigation. The LOFTRAN code family consists of the following versions:

- LOFTRAN – operating plant non-LOCA analyses
- LOFTTR2 – operating plant SGTR analyses
- LOFTRAN – AP-passive plant non-LOCA analyses
- LOFTTR2 – AP-passive plant SGTR analyses

The relationship between the code versions is illustrated in Figure 4-1.

The LOFTTR2 code is a specialized version of the LOFTRAN code modified for the analysis of SGTR events. LOFTTR2 includes an enhanced steam generator secondary side model, a tube rupture break flow model, and improvements to allow simulation of operator actions. This code version is documented in References 2, 3 and 4 and has been reviewed and approved by the NRC for SGTR analyses. LOFTTR2 is currently used for licensing analyses in support of operating plant fuel reloads and plant upgrades.

For non-LOCA events relying on passive safeguards features and SGTR analyses of the AP600, modifications to LOFTRAN and LOFTTR2 were made to simulate the passive plant features. The AP600 is a two-loop pressurized water reactor with passive emergency safeguards features. The passive plant versions of LOFTRAN and LOFTTR2 are referred to as LOFTRAN-AP and LOFTTR2-AP. The principal changes made for the passive plant code versions for design basis analyses consist of adding models for the passive residual heat removal (PRHR) system and the core makeup tanks (CMTs).

A description of the models added to LOFTRAN-AP and LOFTTR2-AP is provided in Revision 1 of WCAP-14234 (Reference 5). Comparisons between tests performed for the AP600 program and LOFTRAN-AP/LOFTTR2-AP are provided in Revision 1 of WCAP-14307 (Reference 6). WCAP-14234 and WCAP-14307 have been reviewed by the NRC and also include NRC review questions and the responses to the questions.

4.2 NRC AP600 LOFTRAN REVIEW

The NRC approved the use of LOFTRAN codes for AP600 analysis in the AP600 FSER, NUREG-1512 (Reference 7). The NRC review of the LOFTRAN codes, summarized in Section 21.6.1 of NUREG-1512, addressed the following areas, which are discussed below:

- Use of auxiliary codes in conjunction with LOFTRAN
- Partial loss of forced RCS flow analysis methodology
- Phenomena Identification and Ranking Table (PIRT)
- Primary and secondary system analytical models in previously approved LOFTRAN versions
- Passive plant components and systems:
  - Automatic Depressurization System (ADS)
  - Core Makeup Tanks (CMTs)
  - Passive Residual Heat Removal (PRHR) Heat Exchanger and In-containment Refueling Water Storage Tank (IRWST)

4.2.1 Use of Auxiliary Codes in Conjunction with LOFTRAN

Transient analyses performed with LOFTRAN are conducted in conjunction with additional support codes. In particular, the FACTRAN code (Reference 8) is used for detailed fuel or heat flux modeling. The THINC (References 9, 10, 11 and 12) or WESTAR (Reference 13) codes were used for Departure for Nucleate Boiling Ratio (DNBR) calculations. These supporting codes were found to be acceptable for AP600 use by the NRC based on previous NRC reviews of these codes and because the fuel design conditions of the AP600 fell within the codes range of validity.

AP1000 analyses will use the FACTRAN support code for detailed heat flux modeling. However an additional support code, VIPRE (Reference 14) will also be used for DNBR calculation. The VIPRE code was developed by Battelle Pacific Northwest Laboratories under sponsorship of the Electric Power Research Institute (Reference 15). VIPRE is widely used throughout the industry and the NRC has given generic approval for its use. The NRC has also reviewed and approved submittals by several utilities for the use of VIPRE for core reload evaluations.

The VIPRE code is flexible and contains input options to permit numerous applications. Like THINC-IV, the VIPRE code is a three-dimensional subchannel thermal-hydraulic code used for describing the reactor core with core boundary conditions supplied by other codes. However VIPRE is also a transient code and temporal variations are calculated. The VIPRE code also includes models of the fuel pin interior comparable to those of FACTRAN for calculation of the transient temperature distribution in a cross section of a fuel rod and the transient heat flux.

Application of the VIPRE code for core thermal-hydraulic analyses by Westinghouse has been previously reviewed and approved for use by the NRC in Reference 14. As described in Reference 14, options selected in the VIPRE code for the Westinghouse methodology give results comparable to those of THINC-IV and FACTRAN codes. The transient core design conditions of the AP1000 non-LOCA analyses are within the validity of the use of the auxiliary codes used in conjunction with LOFTRAN.

4.2.2 Partial Loss of Forced RCS Flow Analysis Methodology

The advanced passive plant designs use reactor coolant systems with two cold legs per reactor coolant loop. The LOFTRAN code simulates only a single cold leg per reactor coolant system (RCS) loop. No changes have been made to the codes to simulate the twin cold leg arrangement. The cold leg arrangement is simulated by lumping the twin cold legs into one. With the lumped cold leg assumption, uniform flow is predicted for the twin cold legs on each RCS loop. This is acceptable for simulation of all events except for those where asymmetric flow conditions are expected. The only events analyzed with LOFTRAN where asymmetric flow conditions within a reactor coolant loop are the following:

- Partial loss of forced reactor coolant flow events
- Locked or broken reactor coolant pump (RCP) shaft events
- Startup of an inactive RCP

Calculation of the net reactor coolant loop flows for use in LOFTRAN is accomplished through the use of auxiliary programs, hand calculations and conservative assumptions. As part of the AP600 licensing effort an outline of the methodology used for calculating conservative transient asymmetric cold leg flows external to LOFTRAN was submitted to the NRC (RAI 440.279 - see Appendix B of WCAP-14234 [Reference 5]). Additionally, sample calculations illustrating the method were also submitted to the NRC (Supplemental Draft Safety Evaluation Report, SDSER Open Item 21.6.1.7-3 - see Appendix B of WCAP-14234).

The NRC concluded that the methodology used for calculation of the effects of a partial loss of flow or locked rotor/broken shaft were conservative. Issues related to simulation of asymmetric cold leg flows were resolved and SDSER Open Item 21.6.1.7-3 was closed.

The reactor coolant loop architecture of the AP1000 is similar to that of the AP600. Twin cold legs and reactor coolant pumps are used in each RCS loop. The conservative approach used for AP600 analyses with asymmetric RCS loop flows is also applicable and acceptable for the AP1000.

4.2.3 Phenomena Identification and Ranking Table (PIRT)

As part of the NRC's review of the AP600, a PIRT was developed for non-LOCA and steam generator tube rupture events. The NRC PIRT was compared to the Westinghouse PIRT submitted in WCAP-14234. The NRC noted that the Westinghouse PIRT was more extensive in depth of coverage of non-LOCA transients. General agreement between the NRC and Westinghouse PIRTs was observed with slight differences.

The NRC PIRT for SGTR ranks the upper head flashing as medium importance while the Westinghouse PIRT ranked the importance of this phenomenon as low. The staff found the differences to be acceptable

because it was noted that calculations indicate that the upper plenum of the vessel stays subcooled with up to 10 ruptured tubes, which is beyond the design basis event.

The NRC PIRT ranked the importance of CMT balance line initial temperature distribution as medium, while the Westinghouse PIRT ranked this phenomenon as low. The differences between the NRC and Westinghouse PIRTs were found to be acceptable because the initial temperature distribution is explicitly input to LOFTRAN and the difference in ranking does not affect the analyses results.

The staff concluded that the PIRT developed for the AP600 transient analyses using LOFTRAN to be applicable and acceptable.

The PIRT developed for the AP600 non-LOCA events was reviewed for applicability to the AP1000. The PIRT review included industry experts and the AP1000 PIRT is presented in Section 2.5 of WCAP-15613 (Reference 16). The basic configuration of the AP1000 is the same as the AP600. AP1000 system and component capacities have been adjusted to accommodate the higher core power rating of AP1000. Due to the similarities of the two designs it is expected that the AP1000 PIRT would be similar to that of the AP600. The review identified no additional phenomena for AP1000 non-LOCA and SGTR analyses. However, the ranking of the CMT "gravity draining injection" phenomenon was changed from "Not Applicable" to medium for steam line and feedwater line ruptures. This is because the pressurizer volume-to-power ratio and the increase in steam generator secondary side volume of the AP1000 could make the RCS more sensitive to shrink and swell events. It was postulated that large enough RCS pressure decreases may occur, such that the CMTs could operate in the gravity drain injection mode rather than the recirculation injection mode. However, this behavior is not expected to occur. A ranking of medium is appropriate at this time until AP1000 analyses confirm that gravity drain CMT injection does not occur during non-LOCA transients.

4.2.4 Primary and Secondary System Analytical Models in Originally Approved LOFTRAN Versions

The NRC approved the original version of LOFTRAN for non-LOCA design basis analyses in 1983 (WCAP-7907-P-A - Reference 1). The NRC approved the specialized steam generator tube rupture code version (LOFTTR2) in WCAP-10698-P-A (References 2). For the AP600, LOFTRAN and LOFTTR2 were modified to include additional models for passive system features. The analytical models in the previously approved versions of LOFTRAN and LOFTTR2 for primary and secondary coolant systems were unchanged for use in the AP600. During its review of AP600, the staff requested additional information on the applicability AP600 thermal-hydraulic conditions to several of the phenomenological models in the previously approved LOFTRAN and LOFTTR2 code versions. The staff concerns included the pressurizer location, wall friction, global pressure location, compressibility effects, reverse flow, and heat transfer options. When the SDSER was issued, Westinghouse had not yet submitted responses to all the staff's RAIs related to the LOFTRAN codes. Submittal of outstanding RAI responses was SDSER Open Item 21.6.1.4-1. Responses to all the outstanding RAIs related to the LOFTRAN codes were completed and submitted to the NRC. Copies of the RAIs and the responses to the NRC were incorporated into Revision 1 of WCAP-14234. The NRC completed its review of these responses and found them to be technically complete and sound, and SDSER Open Item 21.6.1.4-1 was closed. The resolution of the staff concerns on AP600 also apply to AP1000.

4.2.5 Passive Plant Components and Systems

The passive plant designs (AP600 and AP1000) contain features or systems important to the analysis of non-LOCA events that differ from licensed operating Westinghouse plants with active safeguards features. These systems include:

- Automatic Depressurization System
- Core Makeup Tanks
- Passive Residual Heat Removal heat exchanger
- In-containment Refueling Water Storage Tank

Additional models or options to existing models were added to the approved LOFTRAN and LOFTTR2 versions to deal with these passive plant features. The code versions modified to deal with passive plant features were called LOFTRAN-AP and LOFTTR2-AP. These new models were reviewed and approved by the NRC for the AP600. The AP1000 models are based on the approved AP600 models with dimensional input adjustments for the configuration changes.

4.2.6 Automatic Depressurization System

As summarized in NUREG-1512, it was the staff's position that LOFTRAN be restricted from application to analysis involving actuation of the ADS, since the code has not been benchmarked against ADS actuation experiments. ADS actuation involves global two-phase flow behavior for blowdown and LOFTRAN does not have the capability to model this behavior. This was SDSER Open Item 21.6.1.7-5 (see Appendix B of Revision 1 to WCAP-14234).

The Westinghouse response to SDSER Open Item 21.6.1.7-5 noted that the ADS system is not activated to mitigate non-LOCA or steam generator tube rupture events. Therefore, detailed modeling of this system is not required in LOFTRAN. In Section 15.6.1 of the AP600 Design Control Document (DCD), (Reference 17), the results of an inadvertent RCS depressurization are presented. This analysis historically covered the RCS depressurization due to inadvertent opening of pressurizer relief valves. The analyses are short-term analyses that demonstrate that the protection system will detect the depressurization and trip the reactor prior to exceeding DNB limits. For this type of analysis, the most limiting transient is one that will result in the most rapid depressurization of the RCS.

The AP600 DCD Section 15.6.1 included a short-term analysis of the inadvertent opening of an ADS path connected to the pressurizer. Analysis of this type of event was performed with LOFTRAN using assumptions that conservatively maximize the relief from the ADS path under consideration. No credit for ADS piping interactions or interactions with the IRWST that may reduce the rate of RCS depressurization is assumed in the analysis. This results in the maximum rate of RCS depressurization. This is the only analysis performed with LOFTRAN that involves the ADS.

In conclusion, the ADS piping interactions and possible interactions with the IRWST have not been assessed in the LOFTRAN code, since the ADS is not used for mitigation of any transients analyzed with the code. The NRC and Westinghouse agreed that the inadvertent opening of the ADS valves is the only transient that may be analyzed with LOFTRAN in which the ADS plays a part. In this case, the ADS is

treated in the same manner as an open power-operated relief valve, for which LOFTRAN has been found acceptable. Consequently, SDSER Open Item 21.6.1.7-5 was closed.

This approach is planned for use on the AP1000 and continues to provide an acceptable and conservative approach for the AP1000.

4.2.7 Core Makeup Tank

The core makeup tanks provide gravity driven borated coolant injection to the reactor coolant system. The tops of the CMTs are connected to the cold leg by the cold leg balance lines, which have normally open isolation valves. The balance lines maintain the CMTs at the same pressure as the reactor coolant system. Discharge lines connect the bottoms of the CMTs to the reactor vessel. Isolation valves in the discharge lines are normally closed. During normal operation, the CMTs and the connection lines are filled with liquid. When the CMTs are actuated by opening the discharge line valves, the CMTs can operate in two modes, re-circulation injection mode and gravity drain injection mode. During non-LOCA transient events, the CMTs work in the re-circulation injection mode. In non-LOCA events the CMTs provide the emergency boration function for the reactor coolant system. Once activated the CMTs may inject sufficient fluid such that the reactor coolant system is overfilled. This system is important in non-LOCA transients as indicated in the AP1000 PIRT presented in WCAP-15613.

A re-circulation injection mode CMT model was added to LOFTRAN for the AP600 program. The model uses 15 fluid nodes for the tank proper, 3 nodes for the balance line and 8 nodes for the injection line. Heat transfer through the core makeup tank wall is also simulated.

The LOFTRAN CMT model was reviewed by the NRC during the AP600 program. The major NRC issue with the LOFTRAN CMT model revolved around the possibility of steam entering the balance line or fluid flashing within the balance line. The LOFTRAN CMT model is not written for the simulation of two-phase flow transients. This issue (SDSER Open Item 21.6.1.7-4) was resolved by the inclusion in LOFTRAN of a penalty that penalizes the CMT buoyancy head such that natural circulation flow within the CMT is terminated.

The architecture of the AP1000 CMT design is the same as that of the AP600. The AP1000 CMT size has been increased relative to the AP600 and flow control orifices have been modified to increase injection flow. The connection points of the CMT and the number of nodes is hardwired in the LOFTRAN CMT model. However, the dimensional characteristics of the core makeup tanks and the connection lines are provided as input to the code. No changes to the LOFTRAN CMT model are needed to simulate the AP1000 CMT.

Validation of the CMT model of LOFTRAN was conducted by comparing code predictions to the AP600 CMT test facility data. These comparisons are documented in Reference 6. Scaling of the CMT test data for the AP600 was reviewed in Reference 16 and the data was found to be applicable to the AP1000.

4.2.8 Passive Residual Heat Removal (PRHR) Heat Exchanger and In-containment Refueling Water Storage Tank (IRWST)

The PRHR heat exchanger, a C-shaped, down-flow single pass heat exchanger, is submerged in the IRWST. Following depressurization of the RCS, the IRWST also supplies inventory to the RCS by gravity feed injection. This injection function of the IRWST is not used in non-LOCA analyses and is not modeled in LOFTRAN. The PRHR system is used for decay heat removal in non-LOCA analyses and is of high importance in several transient events.

PRHR and IRWST models were added to LOFTRAN for AP600 analyses. The PRHR model can contain up to 45 nodes divided into five regions. Heat exchanger tube nodes may have a horizontal or vertical orientation for buoyancy head and heat transfer calculations. The model transfers heat from the PRHR to the IRWST. The IRWST is simulated as a single homogeneous node. Once the fluid in the IRWST reaches the saturation point then steaming from the IRWST is accounted for.

The LOFTRAN PRHR and IRWST models were reviewed by the NRC during the AP600 program. The principle issues with the LOFTRAN model centered on the inability of the model to calculate thermal stratification within the IRWST if a single homogeneous fluid region model is used and the selection of the appropriate pool boiling heat transfer coefficient used on the outside of PRHR tubes.

The NRC questioned the validity of using a homogeneous, mixed condition in the IRWST when temperature stratification is likely. This issue was resolved by performing sensitivity studies with the LOFTRAN model using temperature stratification profiles from the SPES and PRHR test programs and demonstrating that using a homogenous IRWST temperature produces conservative non-LOCA transient analysis results.

The correlation used for pool boiling in the LOFTRAN PRHR model was developed from the Westinghouse PRHR test program. The PRHR test program used a configuration with three straight tubes. The NRC questioned the validity of these tests for defining the heat transfer of the PRHR. This issue was resolved based on comparisons of the LOFTRAN PRHR model to other tests. LOFTRAN simulations of SGTR tests at the SPES-2 facility were performed. The PRHR performance during these tests was accurately predicted by LOFTRAN. Westinghouse performed further blind test analyses of the PRHR heat transfer by calculating the performance of the full height C-tube heat exchanger used in the ROSA AP600 confirmatory tests. The analyses of the ROSA tests indicated the heat transfer correlation used in the LOFTRAN model conservatively predicted the heat transfer measured in the experiment.

The architecture of AP1000 PRHR design is the same as that of the AP600. The AP1000 uses a larger heat exchanger and the inlet and outlet piping sizes of the AP1000 have been increased. The architecture of the PRHR model is hardwired in LOFTRAN. However, the dimensional characteristics of the PRHR are set as input and can be adapted for the increased size. The acceptable resolution of NRC concerns on AP600 applies to the AP1000.

4.3 CODE VERSIONS FOR AP1000 ANALYSES

The AP600 non-LOCA analyses were performed using Version 1.8 of LOFTRAN-AP and the steam generator tube rupture analysis was performed using Version 1.6 of LOFTTR2-AP. The advanced plant

code versions were developed by adding passive system features to the licensed operating plant analysis versions of LOFTRAN and LOFTTR2 available during the AP600 program.

Enhancements and upgrades to the LOFTRAN version used for operating plants have continued independent of the AP600 and AP1000 passive plant programs. The principal upgrades to the operating plant LOFTRAN version includes the following:

- Data transfer interfaces to other auxiliary computer codes
- Enhanced pressurizer safety and relief valve models
- Enhanced secondary side safety and relief valve models
- Input and output formatting
- VVER system models
- Enhanced RCS thick metal heat transfer model (description submitted to the NRC as Supplement 1 of WCAP-7907-S1 (Reference 18)

As part of the AP1000 project, the LOFTRAN-AP code will be upgraded to be consistent with the LOFTRAN version used for operating PWRs. Many of the enhancements to the operating plant version of LOFTRAN are not applicable to the passive plant analyses and therefore will not be used. Two of the upgrades that will be incorporated and used in the passive plant code include the data transfer interfaces to auxiliary computer codes, and the enhanced pressurizer and secondary side relief valve models. The data transfer interfaces to auxiliary computer codes upgrade will allow data to be transferred to auxiliary codes such as FACTRAN or the core subchannel DNBR analysis codes such as VIPRE. The enhanced pressurizer and secondary side relief valve models, which use more detailed models to allow individual valve inputs rather than a lumped valve model, improves the realism of the relief characteristics and aids in evaluating the cycling processes of the safety valves. The realistic safety valve model was used in supporting analyses submitted to the staff in response to RAIs during the AP600 Design Certification review. As shown in the analyses, inclusion of this model results in an increase in the pressurizer level swell, and therefore, tends to reduce the predicted margin to pressurizer overfill for transient events when safety valve opening is predicted. The staff review for AP600 included review of the model. The new LOFTRAN-AP version is consistent with the LOFTRAN code version currently in use for analysis support of operating PWRs. Models for the passive system which are approved for the AP600 remain unchanged and are applicable for the AP1000.

Many of the non-LOCA analyses do not rely on passive system features for mitigation of the events. These events behave in a similar manner as licensed operating PWRs and can be analyzed using the same versions of LOFTRAN as operating plants. Table 4-1 summarizes the transients analyzed using the LOFTRAN code family and identifies which code versions can be used for the analyses. The AP600 or AP1000 results of those transients that can be analyzed using either the operating plant version or the passive plant code version are the same independent of the version used. The passive plant models in LOFTRAN approved during the AP600 Design Certification review are the same models that will be used for AP1000. Other changes associated with the transitions to the latest revisions of LOFTRAN have been

approved for operating plants and are acceptable for AP1000 because they do not affect the models associated with the passive features.

4.4 CONCLUSIONS

NRC review of the LOFTRAN codes was performed for the AP600. In NUREG-1512, the staff concluded that LOFTRAN had been modified to include the necessary models for the AP600 plant features and behavior expected during non-LOCA transients and was acceptable for the AP600 passive reactor design.

Preliminary AP1000 analyses were performed for selected non-LOCA and SGTR design basis events using the methods and LOFTRAN versions validated for the AP600. The results of these AP1000 analyses are presented in WCAP-15612 (Reference 19). The results of the AP1000 analyses showed safety margins comparable to those of the AP600 and resulted in no new phenomena or significant differences in plant performance characteristics.

AP1000 PIRT and scaling assessments are summarized in WCAP-15613. The results of the preliminary AP1000 analyses indicate that non-LOCA and SGTR transients for passive plants are similar to conventional operating PWRs with the exception of the PRHR heat exchanger and the CMT injection models. Models for the PRHR heat exchanger and the core make up tanks were incorporated into LOFTRAN for the AP600 project. As the PIRT and scaling of these two effects are similar for AP600 and AP1000, analysis codes that acceptably predict AP600 performance will acceptably predict AP1000 performance.

The basic configuration of the systems and components of the AP1000 remains the same as that of the AP600. The capacities of AP1000 systems and components have been adjusted to accommodate the higher core power of the AP1000 relative to the AP600. With respect to systems and components important to non-LOCA and steam generator tube rupture safety analyses, the general configuration of the AP1000 is the same as that approved for the AP600. While the architecture of the models needed for safety analyses within the LOFTRAN code are hardwired, the geometric dimensions are set by input parameters and will be modified for the AP1000 analyses without modifications to the computer code. In applying the LOFTRAN code family to the AP1000 analyses, conservative treatments for input parameters will be applied consistent with the analyses performed for the AP600 and operating plants. This includes the use of uncertainties on initial conditions, the use of upper and lower bound core reactivity coefficients, bounding protection system setpoints and actuation delays, and bounding performance parameters for emergency safeguards systems such as the PRHR and CMTs. The selection of the upper or lower bound input values is established on an event-by-event basis to produce conservative results with respect to acceptance criteria.

4.5 REFERENCES

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18. WCAP-7907-S1-P, "LOFTRAN Code Description, Supplement 1 - LOFTRAN Thick Metal Mass Heat Transfer Models," January 2001.

19. WCAP-15612, "AP1000 Plant Description and Analysis Report," December 2000.

Table 4-1 Applicable Code Versions for Passive Plant Design Basis Analysis

Events	SAR Section	Code Version		
		LOFTRAN	LOFTRAN-AP	LOFTTR2-AP
Feedwater system Malfunction that Result in a Decrease in feedwater Temperature or an Increase in Feedwater flow	15.1.1 15.1.2	X	X	
Excessive Increase in Secondary Steam Flow	15.1.3	X	X	
Inadvertent Opening of a Steam Generator Relief or Safety Valve and Steam System Piping Failure	15.1.4 15.1.5		X	
Inadvertent Operation of the PRHR	15.1.6		X	
Loss of External Electrical Load Turbine Trip Inadvertent Closure of Main Steam Isolation Valves Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip	15.2.2 15.2.3 15.2.4 15.2.5	X	X	
Loss of ac Power to Plant Auxiliaries Loss of Normal Feedwater Flow	15.2.6 15.2.7		X	
Feedwater System Pipe Breaks	15.2.8		X	
Partial Loss of RCS Flow Complete Loss of RCS Flow	15.3.1 15.3.2	X	X	
RCP Pump Shaft Seizure RCP Pump Shaft Break	15.3.3 15.3.4	X	X	
Uncontrolled RCCA Bank Withdrawal at Power	15.4.2	X	X	
Startup of an Inactive RCP at an Incorrect Temperature	15.4.4	X	X	
Inadvertent Operation of the CMT During Power Operation	15.5.1		X	
Chemical and Volume Control system Malfunction that Increase Reactor Coolant Inventory	15.5.2		X	
Inadvertent Opening of a Pressurizer Relief Valve or Inadvertent Opening of an ADS Valve	15.6.1		X	
Steam Generator Tube Rupture	15.6.3			X

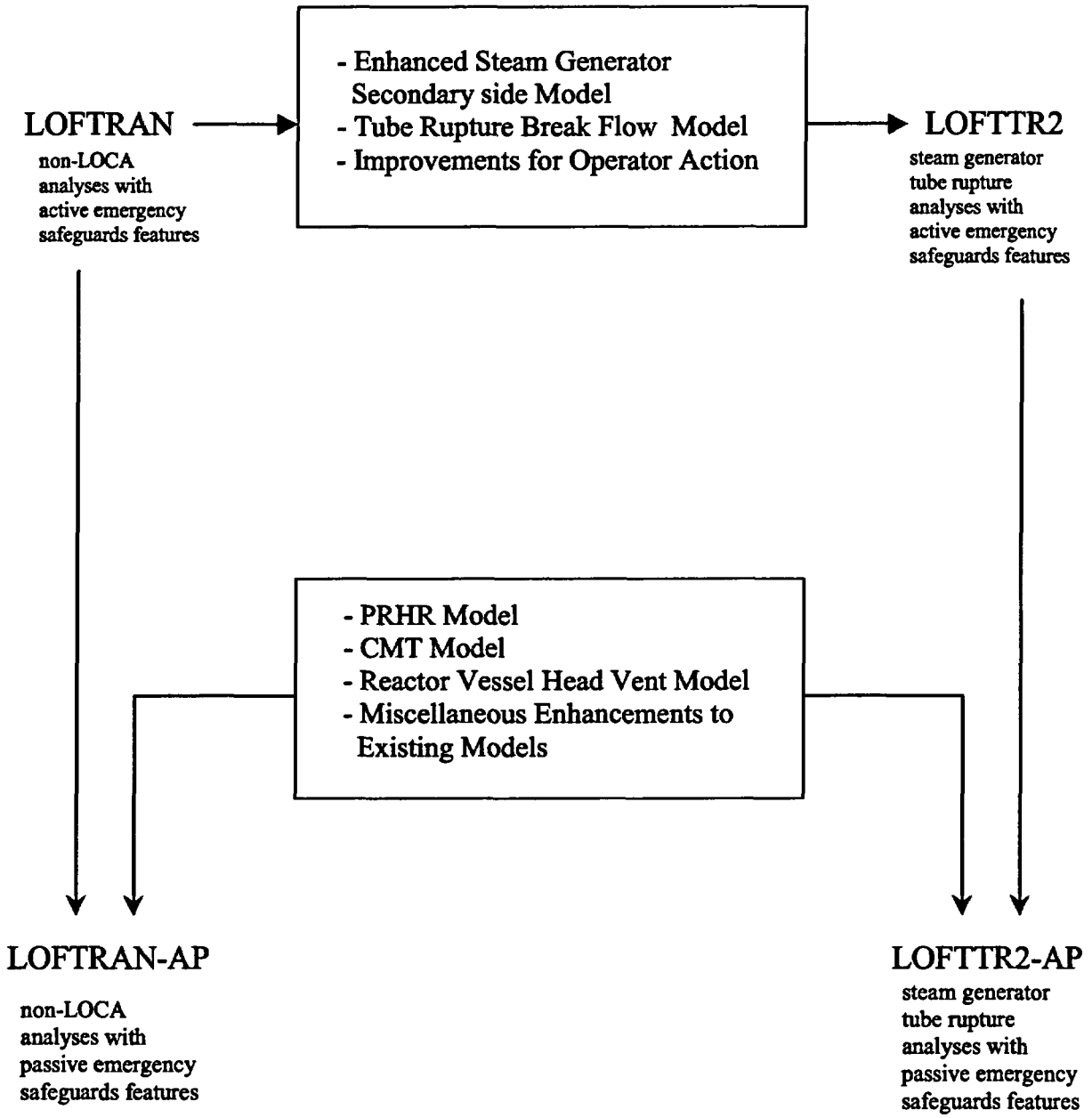

Figure 4-1 Relationship of LOFTRAN Code Versions

5.0 APPLICABILITY OF WGOTHIC FOR AP1000 CONTAINMENT INTEGRITY ANALYSES

5.1 BACKGROUND

The GOTHIC code is a state-of-the-art program for modeling multi-phase flow. The GOTHIC code was developed over a period of time from other qualified thermal-hydraulic computer codes as shown in Figure 5-1.

GOTHIC consists of three separate programs, the preprocessor, solver, and postprocessor. The preprocessor allows the user to rapidly create and modify an input model. The solver performs the numerical solution for the problem. The postprocessor, in conjunction with the preprocessor, allows the user to rapidly create graphic and tabular outputs for most parameters in the model.

The GOTHIC solver program calculates the solution for the integral form of the conservation equations for mass, momentum, and energy for multi-component, two-phase flow. The conservation equations are solved for three fields: continuous liquid, liquid drops, and the steam/gas phase. The three fields may be in thermal nonequilibrium within the same computational cell. This allows the modeling of subcooled drops (for example, containment spray) falling through an atmosphere of saturated steam. The gas component of the steam/gas field can be comprised of up to eight different noncondensable gases with mass balances performed for each component. Relative velocities are calculated for each field, as well as the effects of two-phase slip on pressure drop. Heat transfer between the phases, surfaces, and the fluid are also allowed.

The GOTHIC solver program is capable of performing calculations in three modes. A model can be created in the lumped-parameter nodal-network mode, the two-dimensional distributed parameter mode, or the three-dimensional distributed parameter mode. Each of these modes may be used within the same model. The lumped parameter nodal-network mode is used for the AP600 containment Evaluation Model.

The GOTHIC code also contains the options to model a large number of structures and components. These include, but are not limited to, heated and unheated conductors, pumps, fans, a variety of heat exchangers, and ice condensers. These components can be coupled to represent the various systems found in any typical containment.

The GOTHIC code has an extensive validation history which was an important consideration in the selection of the code for further development for modeling of the PCS. The GOTHIC code validation program includes both a comparison of code-calculated results with analytical solutions to specified standard problems and a comparison of code-calculated results with experimental data. The results of the EPRI-sponsored GOTHIC code validation program are presented in Reference 1, Enclosure 1. Table 5-1 lists some of the tests used in the GOTHIC code validation program. The phenomenological models validated by each test are cross-referenced and presented in Table 5-2. In addition, industry experience using GOTHIC in the lumped parameter mode, as well as attempts to improve results using multi-dimensional analyses, are described in WCAP-15846, Rev. 0, (Reference 2) Appendix 9.C.3.

After reviewing the qualifications of the available containment analysis codes, Westinghouse selected and purchased the GOTHIC code for further development and application to modeling of the AP600 passive containment design. Westinghouse developed special subroutines to mechanistically calculate the heat and mass transfer and to track the liquid films for the passive containment cooling system (PCS). These subroutines were incorporated into GOTHIC Version 4.0 to create WGOTHIC Version 4.2. See WCAP-15846 (Reference 2), Sections 3.3–3.5 for a detailed description of the Westinghouse Clime Model.

The GOTHIC Version 4.0 validation test problems were re-run with WGOTHIC to determine if any of the changes that were made to incorporate the PCS heat and mass transfer models would affect the validation results - they did not. The WGOTHIC PCS heat and mass transfer models were validated by comparison with various separate effects tests as listed in Table 5-3. The results of this comparison are documented in WCAP-14326 (Reference 3).

Both lumped parameter and distributed parameter (3-D) models of the large-scale test facility were constructed with WGOTHIC for validation of the passive containment, evaluation model methodology. The "well mixed" assumption, implicit in the lumped parameter modeling approach, in combination with the neglect of the velocity component for the internal condensation heat and mass transfer, resulted in the lumped parameter model significantly overpredicting the system pressure in the LST facility. A more complete description of the validation models and results of the comparison are presented in WCAP-14382 (Reference 4).

The WGOTHIC AP600 containment evaluation model makes use of the lumped parameter modeling approach. The WGOTHIC AP600 containment evaluation model is a complicated structure consisting of a large number of lumped parameter volumes, some of which contain heat sinks and/or PCS clime components. The lumped parameter volumes are connected with flow paths. Boundary conditions are used to supply the transient mass and energy release from the break source. A complete description of the AP600 containment evaluation model is provided in Section 4.0 of WCAP-15846.

The lumped parameter modeling approach is based on 30 years of nuclear industry experience. The industry experience has identified limitations and biases in the lumped parameter modeling approach that are due primarily to the oversimplification of the momentum formulation. These limitations and biases were identified based on model comparisons to international tests at different scales.

Several limitations and biases were applied to models for important phenomena in the WGOTHIC AP600 containment evaluation model to develop a bounding methodology for calculating the containment pressure. The WGOTHIC AP600 containment evaluation model limitations and biases includes:

- The use of lower bound multipliers on the heat and mass transfer correlations to reduce condensation and evaporation on the PCS,
- The use of only the free convection correlation (no forced convection component is allowed) to calculate the condensation heat and mass transfer to the inside surface of the shell,
- A 10-percent reduction of the containment shell emissivity input value,

- The use of the maximum Passive Containment Cooling System Water Storage Tank (PCCWST) water temperature allowed by the Technical Specifications to minimize sensible heat transfer to the applied liquid film,

- The use of an "evaporation limited" PCS water flow rate to minimize sensible heat transfer to the applied liquid film,

- The assumption of a single failure of one of two PCS cooling water flow control valves, along with the assumption of the minimum initial PCCWST water inventory allowed by the Technical Specifications to minimize the initial PCS water flow rate,

- The use of a 337 second delay time to establish the steady state external film coverage and initiate evaporation heat and mass transfer from the shell,

- The use of a PCS annulus loss coefficient that is 30-percent larger than the value measured in the test program to minimize the air flow rate and evaporation from the shell,

- The use of the maximum containment internal air temperature and pressure allowed by the Technical Specifications as the model initial conditions,

- The use of an initial zero-percent relative humidity to maximize the internal stored energy inside containment

- The elimination of compartment floors as potential heat sinks,

- The elimination of heat transfer to conductors within dead-ended volumes after blowdown, and

- The use of a 20-mil air gap between the steel and concrete on jacketed heat sinks,

5.2 RESOLUTION OF MAJOR ISSUES

Before accepting the WGOTHIC AP600 containment evaluation model, the NRC and ACRS identified several issues that had to be resolved. The three main issues were:

- modeling circulation and mixing within the containment (requires justification for the use of lumped-parameter noding),

- modeling the Passive Containment Cooling System (PCS) condensation and evaporation heat removal (requires justification for the water coverage input and the clime heat and mass transfer models), and

- validation of the WGOTHIC AP600 containment evaluation model (requires justification for the use of the LST and other test facilities).

Westinghouse provided documentation (Section 9 of WCAP-15846, Rev. 0) to support the use of lumped parameter noding to model circulation and mixing in the WGOTHIC AP600 containment evaluation model. Experimental results from various international tests were examined for applicability to loss-of-coolant (LOCA) and main steam line break (MSLB) events in a passive (externally cooled) containment design. Assuming an initially well-mixed atmosphere within the facility, the tests showed global circulation would occur when the break source was located in a lower compartment and there were relatively large openings between interconnected compartments (similar to a LOCA within the AP600). In addition, the Large Scale Test (LST) and Heissdampfreaktor (HDR) tests also showed that circulation and mixing were enhanced after the application of external cooling water to the top of the test facility. Steam condensing at the top inside surface of the test facility resulted in negatively buoyant plumes of cooler air falling downward, increasing the global circulation and mixing within the test facility.

The passive containment structure employed by AP600 and AP1000 was designed to promote global circulation following a LOCA event. There are large openings between compartments to minimize flow restrictions. The RCS piping is located in the lower compartments; this maximizes the driving force for global circulation by the buoyant steam plume. Finally, the PCS water is applied at the top and flows down along the containment shell; this maximizes the driving force for global circulation by the negatively buoyant plumes generated by condensation on the inside surface of the containment shell.

Due to the break location, some of the lower compartments within the passive containment may not be as strongly affected by the naturally-induced global circulation as others. To account for the potential effect of stratification within compartments of the WGOTHIC AP600 containment evaluation model, heat transfer to floors is eliminated and, after blowdown is complete, heat transfer to conductors within dead-ended compartments is turned off.

Westinghouse provided documentation (WCAP-14326 [Reference 3]) to support the use of the heat and mass transfer correlations for condensation and evaporation in the WGOTHIC AP600 containment evaluation model. Data from separate effects heat and mass transfer tests were used to validate the correlations. The range of the independent dimensionless parameters from the tests covered the operating range of the AP600. Bounding multipliers (0.73 for condensation and 0.84 for evaporation) were used to conservatively bound (reduce) heat and mass transfer in the WGOTHIC AP600 containment evaluation model.

Westinghouse provided documentation (Section 7 of WCAP-15846, Rev. 0) to support the PCS water coverage model in the WGOTHIC AP600 containment evaluation model. Test data from a full-scale section of the containment dome was used to determine the initial water coverage fraction input values for the AP600 containment evaluation model. The time-dependent water flow rate input for the AP600 containment evaluation model was limited to either the actual PCS flow rate (assuming a failure of one of two parallel valves to open) or the conservatively estimated transient evaporation rate, whichever is smaller. This conservatively ignores the effect of sensible heating of the runoff flow rate.

The standard set of GOTHIC code qualification tests includes comparisons with data from a number of different test facilities to validate the code and lumped parameter modeling technique. This same set of tests was run with WGOTHIC. The results of these tests confirmed that the changes Westinghouse made to the software had no effect on the results of the GOTHIC code qualification.

Westinghouse provided documentation (WCAP-14845 [Reference 5], Section 10.2) to support the use of steady state test data from the LST to validate the WGOTHIC code and AP600 containment evaluation model. Problems with the design and scaling of the LST facility limited its usefulness for AP600 transient comparisons, however, the steady-state data was determined to be acceptable for validating the heat and mass transfer correlations as well as providing comparison points during the slowly changing long term cooling transient. The calculated results from a lumped parameter model of the LST facility were compared to the test data (WCAP-14382 [Reference 3] and WCAP-14967 [Reference 6]) to support the WGOTHIC AP600 containment evaluation model. The lumped parameter model calculated a pressure response that was conservative (higher) relative to the test data.

The NRC received this information and reviewed it using a process similar to the one that is outlined in the current Draft Standard Review Plan Section 15.0.2 of NUREG-0800 and the Draft Regulatory Guide, DG-1096. After completing a thorough review of this information, the NRC determined that the WGOTHIC computer program, combined with the conservatively biased AP600 containment evaluation model, could be used to demonstrate that the AP600 containment design meets the requirements of General Design Criteria (GDCs) 16, 38 and 50. This approval was subject to the limitations and restrictions described in Section 5.1 and listed in subsection 21.6.5.8.3 of NUREG-1512, AP600 Final Safety Evaluation Report (FSER) (Reference 7).

With regard to the modeling of circulation and mixing for the LOCA event, the AP600 FSER states: "Initially, the DBA blowdown and PCS operation generate a nearly homogeneous distribution of steam and non-condensable gases. In the longer term, the actuation of the fourth stage automatic depressurization system valves (ADS-4), at approximately 1000 seconds, supports a circulation pattern which tends to sustain the homogeneity of the containment atmosphere. Under these conditions, the lumped parameter representation is acceptable for evaluating the AP600 peak containment pressure." With regard to the modeling of circulation and mixing for the MSLB event, the AP600 FSER states: "The degree of homogenization is a strong function of break location, direction, and momentum. The MSLB blowdown creates circulation patterns that tend to homogenize the containment atmosphere above the break location sufficiently to accept the lumped-parameter representation for the evaluation of the AP600 peak containment pressure."

With regard to the PCS heat and mass transfer correlations, the AP600 FSER states: "The staff was concerned with uncertainties in the correlations and the data base, and Westinghouse has biased the correlations to account for these uncertainties. Based on comparisons of the predicted-to-measured Sherwood numbers, the bias for the evaporation mass transfer is a multiplier of 0.84 on the correlations. For condensation, the bias multiplier is 0.73 on the mass transfer correlations. The same multipliers are applied to the heat transfer correlations, based on the mass and heat transfer analogy. The multipliers were chosen to bound the comparisons and are acceptable."

With regard to validation testing, the AP600 FSER states: "The staff concludes that the evaluation model contains sufficient conservatism, including factors to compensate for shortcomings in the LST, to accept WGOTHIC in combination with the AP600 evaluation model for DBA licensing analyses to support design certification."

5.3 JUSTIFICATION FOR THE USE OF THE WGOTHIC CODE AND AP600 CONTAINMENT EVALUATION MODEL METHODOLOGY FOR APPLICATION TO THE AP1000

Both the AP1000 and AP600 employ a Passive Containment Cooling System. The AP1000 containment structure is taller, but maintains the same diameter and internal layout as the AP600. A detailed comparison of the AP600 and AP1000 plant designs is provided in WCAP-15612 (Reference 8).

The capability requirements for the AP1000 containment evaluation model are the same as AP600. To be able to model the passive containment cooling system, the evaluation model must be able to model:

- The transport of break mass and energy (steam) to the containment shell,
- The condensation of steam on the inside surface of the containment shell,
- The transport of the condensate film on the inside surface of the containment shell,
- The conduction of heat through the containment shell,
- The transport and heating of the applied liquid film on the outside surface of the containment shell,
- Evaporation from the applied liquid film on the outside surface of the containment shell and,
- The natural draft cooling air flowing through the downcomer, riser and chimney of the shield building.

As described earlier, Westinghouse developed special subroutines to mechanistically calculate the heat and mass transfer and to track the liquid films for the passive containment cooling system. These subroutines were appended to the GOTHIC Version 4.0 code to create WGOTHIC Version 4.2.

To determine the applicability of using the WGOTHIC code (Version 4.2) and AP600 containment evaluation model methodology for performing the AP1000 containment DBA analyses, Westinghouse performed the following:

- Reviewed the AP600 containment PIRT for application to the AP1000,
- Reviewed the AP600 containment scaling analysis for application to the AP1000 and,
- Compared the test data ranges of the important dimensionless parameters for heat and mass transfer and water coverage with the operating range for the AP1000.

The AP600 containment PIRT was reviewed to determine if there were any new phenomena or any change in the importance ranking of the existing phenomena with respect to the AP1000 containment and RCS design changes. This review was documented in WCAP-15613 [Reference 9], Section 2.6. No new

phenomena were identified and there were no significant changes in the ranking of phenomena as a result of the AP1000 design changes.

An LST scaling assessment was performed for AP1000 and compared with AP600 (see WCAP-15613, Section 4.2). Due to its relatively low and constant steam injection flow rate, the LST was not well scaled to model the blowdown transient response for either AP600 or AP1000. However, the phenomena were well scaled in the quasi-steady state phase. Therefore, the steady state LST data were determined to be acceptable for use as a source of separate effects test data for internal condensation, above-deck steam distribution, external heat transfer, and external water coverage.

The ranges of the dimensionless parameters for the heat and mass transfer correlations were examined to determine if the existing test data covered the AP1000 operating range (see WCAP-15613, Section 4.2). The test data covered the upper range of the AP1000 dimensionless parameters for the heat and mass transfer correlations in the important riser region of the annulus. Therefore, the correlations are also considered to be valid for the AP1000 containment evaluation model.

Experimental test data and correlations were reviewed to determine if the increase in containment height would affect the thermally-induced mixing within the open volume above the operating deck. Both the correlations and test data suggest that increasing the containment height would increase the turbulence and improve the mixing (see WCAP-15846, Section 9C).

An alternate analysis methodology was used to independently assess the relative degree of mixing in the open volume above the operating deck for the AP600 and AP1000. Detailed, 2-dimensional slice Computational Fluid Dynamics (CFD) models representing this region were constructed for both the AP600 and the AP1000 (see WCAP-15613, Section 4.2). The flow and velocity patterns for the AP600 and AP1000 were very similar. Both models predicted cold falling plumes near the walls and a hot rising plume near the center of the volume. Except for the small boundary layers very close to the walls and within the central plume, the temperature profile within the volume was nearly uniform. Therefore, based on the experimental test data, correlations, and results from the alternate analysis approach, the well-mixed assumption for this region was also considered to be valid for the AP1000 containment evaluation model.

The operating ranges of the liquid film coverage parameters for AP600 and AP1000 were compared to the composite PCS test data. The test data covered the operating range of the important film coverage parameters (minimum film Reynolds number and maximum heat flux) for both AP600 and AP1000. Therefore, the constant coverage area input values and the model for calculating the evaporation-limited PCS water flow rate input that was used for AP600 are also applicable to the AP1000.

In summary, both the AP600 and AP1000 employ the same passive containment cooling system design features so the events and phenomena to be analyzed in the AP1000 containment evaluation model are the same as the AP600. The range of important dimensionless parameters from the PCS test data covers the operating range of both the AP600 and AP1000, so the WGOTHIC heat and mass transfer correlations remain acceptable. Since the containment designs are similar and since the heat and mass transfer correlations remain acceptable. WGOTHIC source code changes are not required for the AP1000 containment evaluation model. The AP1000 containment evaluation model will use the same bounding methodology that was accepted by the NRC for the AP600.

5.4 CONCLUSIONS

The bounding WGOTHIC AP600 containment evaluation model was accepted by the NRC to demonstrate that the AP600 containment design meets the requirements of GDCs 16, 38, and 50 (subject to the limitations and restrictions listed in Section 21.6.5.8.3 of the AP600 FSER). Both the AP600 and AP1000 employ the same passive containment cooling system design features so the events and phenomena to be analyzed in the AP1000 containment evaluation model are the same as AP600. To justify the use of the WGOTHIC and the AP600 containment evaluation model for application to the AP1000. Westinghouse provided documentation to demonstrate that:

- The AP1000 containment PIRT is unchanged from the AP600.
- The AP1000 operating range of the important dimensionless parameters for heat and mass transfer and liquid film coverage are bounded by the existing test data.
- The experimental test data, correlations, and alternate analysis methodology confirm the volume above the AP1000 operating deck is also sufficiently mixed to allow the use of the lumped parameter modeling approach.

Therefore, Westinghouse intends to use the previously accepted, bounding AP600 containment evaluation model, which is based on WGOTHIC version 4.2 to perform the AP1000 containment DBA analyses with appropriate input modifications to reflect the AP1000 containment design changes.

5.5 REFERENCES

1. Westinghouse Letter NTD-NRC-95-4563, "GOTHIC Version 4.0 Documentation," B. A. McIntyre to Quay (NRC), September 21, 1995.

2. WCAP-15846, Rev. 0, "WGOTHIC Application to AP600 and AP1000," J. Woodcock et al., April 2002.

3. WCAP-14326, "Experimental Basis for the AP600 Containment Vessel Heat and Mass Transfer Correlations," F. Delose, et al., April 1998.

4. WCAP-14382, "WGOTHIC Code Description and Validation," M. Kennedy et al., May 1995.

5. WCAP-14845, Rev. 3, "Scaling Analysis for AP600 Containment Pressure During Design Basis Accidents," D. R. Spencer, et al., March 1998.

6. WCAP-14967, "Assessment of Effects of WGOTHIC Solver Upgrade from Version 1.2 to 4.1," September 1997.

7. NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998.

8. WCAP-15612, "AP1000 Plant Description and Analysis Report," M. M. Corletti et al., December 2000.

9. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

Table 5-1 GOTHIC Validation Tests	
Battelle-Frankfurt Tests D-1, D-15, D-16 (BFMC)	Modeling: 7 lumped parameter volumes, junctions Phenomena: Blowdown transients, subcompartment pressurization, wall differential pressures
Battelle-Frankfurt Test 6 (BFMC)	Modeling: 1 distributed parameter volume (55 cells), conductors, junctions Phenomena: Hydrogen transport by convection and diffusion
Battelle-Frankfurt Tests 12, 20 (BFMC)	Modeling: Combination of 5 lumped and 1 distributed parameter volumes (2 cells), conductors, junctions Phenomena: Hydrogen transport by convection and diffusion
Battelle-Frankfurt Tests C-13, C-15 (BFMC)	Modeling: 10 lumped parameter volumes, conductors, junctions Phenomena: Main steamline break, pressure/temperature response
Hanford Engineering Development Laboratory Tests HM-5, HM-6 (HEDL)	Modeling: 1 distributed parameter volume (300 cells), conductors, junctions Phenomena: Hydrogen mixing in a large, simulated containment
Light Water Reactor Aerosol Containment Experiments Tests LA-5, LA-6 (LACE)	Modeling: Combination of 1 lumped and 1 distributed parameter (2 cells) volumes, conductors, junctions Phenomena: Severe accident response to sudden containment failure
Marviken Full-Scale Containment Tests 17, 24 (MARV)	Modeling: 21 lumped parameter volumes, conductors, junctions Phenomena: Pressurized high temperature steam blowdown
Carolina's Virginia Tube Reactor Tests 3, 4, 5 (CVTR)	Modeling: 2 lumped volume and a 2 distributed parameter volume (20 cells) models, conductors, junctions Phenomena: Steam blowdowns (T31.5 includes hydrogen/helium)
Heissdampfreaktor Tests V21.1, T31.1, T31.5, V44 (HDR)	Modeling: 37 lumped parameter volumes, conductors, junctions Phenomena: Steam blowdowns (T31.5 includes hydrogen/helium)

Table 5-2 GOTHIC Phenomenological Models Validated by Test

Item	BFMC	HEDL	LACE	MARV	CVTR	HDR
Fluid momentum	X		X	X		
Energy transport	X		X	X		
Noncondensable gases	X	X	X	X	X	X
Equations of state	X		X	X		
Pressure response	X	X	X	X	X	X
Temperature response	X	X	X	X	X	X
Humidity response	X	X	X	X	X	X
Hydrogen transport	X					
Energy sources	X	X	X		X	X
Subcompartment analysis	X			X		
High energy line breaks	X					
PWR standard containment			X			
BWR pressure suppression				X		
Fluid/structure interaction	X					
Conductors	X					
Subdivided volumes	X					
Turbulence	X					
3-D calculations	X	X		X		

Table 5-3 WGOTHIC PCS Heat and Mass Transfer Model Validation	
STC Dry Flat Plate	Forced convection heat transfer, heated flat plate in channel-type geometry
Westinghouse Large Scale Test – Dry External Heat Transfer	Mixed convection heat transfer, 1/8-scale AP600 containment, internally steam heated, externally cooled by air
Hugot Heated Channel Tests	Mixed convection heat transfer, isothermal parallel plates in channel-type geometry
Eckert and Diaguila Tests	Mixed convection heat transfer, externally steam heated tube
Siegel and Norris Tests	Mixed convection heat transfer, parallel vertical flat plates in channel-type geometry, constant heat flux
STC Wet Flat Plate	Forced convection evaporation heat and mass transfer, heated flat plate in channel-type geometry
Gilliland and Sherwood Evaporation Tests	Mixed convection evaporation heat and mass transfer from the inside surface of a vertical heated pipe
University of Wisconsin Condensation	Forced convection condensation heat and mass transfer in channel-type geometry
Westinghouse Large Scale Test – Internal Condensation	Free convection condensation heat and mass transfer, 1/8-scale AP600 containment, internally steam heated, externally cooled by evaporation

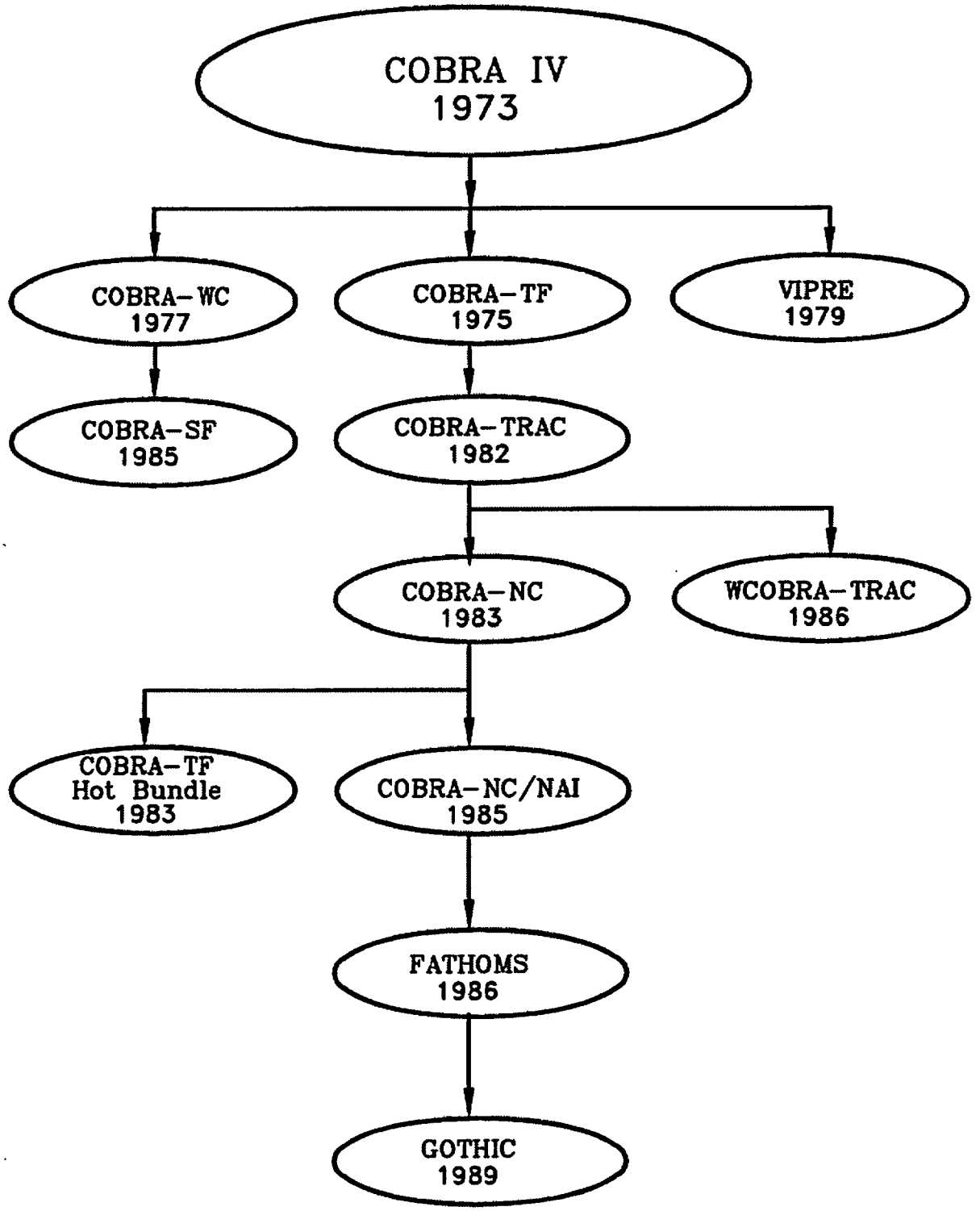

Figure 5-1 Summary of GOTHIC Historical Development

6.0 CONCLUSIONS

This report provides an assessment of the analysis codes that were developed and approved for the AP600 Design Certification to determine their applicability and use for Design Certification of an AP1000. The analysis codes that were approved for the purposes of performing safety analyses of the AP600 passive plant are:

- LOFTRAN – transient analyses
- NOTRUMP – small-break LOCA analysis
- WCOBRA/TRAC – large break LOCA and long-term cooling analysis
- WGOTHIC – containment analysis

This report describes the basis for the use of these safety analysis codes approved for a plant design with passive safety features for a Design Certification of an AP1000. For each of the thermal-hydraulic analysis codes, the report discusses the basis for that approval as described in NUREG-1512, Final Safety Evaluation Report (FSER) Related to Certification of the AP600 Standard Design (AP600 FSER, Reference 1). This report discusses the basis for their approval for AP600, and provides an assessment as to how that basis can be applied to AP1000. In addition, the main attributes associated with the graded approach to assessment and application of an evaluation model outlined in Draft Regulatory Guide DG-1096 are addressed for each analysis code.

In this report, our bases for the use of the analysis codes (previously validated and approved for AP600) is described. For each of the thermal-hydraulic analysis codes that were developed and approved as part of AP600 Design Certification, LOFTRAN, NOTRUMP, WCOBRA/TRAC, and WGOTHIC, the report discusses the basis for that approval as described in the AP600 FSER. A summary of the major issues for each code is provided with a discussion of the applicability of the AP600 code approval basis to the AP1000. This provides the justification for the continued use of these approved codes for AP1000.

The following summarizes the conclusions of this report specific to each code:

1. The LOFTRAN-AP code that was approved for AP600 can be used for the purposes of performing conservative analysis of the transient events presented in Chapter 15 for AP1000. The basis for this conclusion is that when considering transient events, no new phenomenon is identified for AP1000 (when compared to AP600). Analysis show that passive plants behave similarly to operating plants with regards to transient events. The test database that supported validation of this code for AP600 is applicable to AP1000. The means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000.

   The main code-related issues identified during the AP600 Design Certification review include:

   - Modeling of asymmetric flow conditions
   - ADS flow for event involving inadvertent opening of ADS valves
   - Impact of flashing or steam in the CMT balance line on CMT flow
   - Impact of thermal stratification in IRWST on PRHR heat transfer

Modeling of asymmetric flow conditions is accomplished through a methodology which employed the use of auxiliary calculations approved for AP600. This methodology can be applied for AP1000. Therefore, the means for resolution is applicable to AP1000.

The ADS flow for inadvertent ADS actuation events was treated in the same manner as an open PORV for which LOFTRAN was found acceptable. This same approach will be used for AP1000. The means for resolution is applicable to AP1000.

The LOFTRAN CMT model is not written for simulation of two-phase flow transients. The possibility of flashing or steam in the CMT balance line was resolved by inclusion of a penalty on the CMT gravity head such that natural circulation flow the CMT flow is terminated. This penalty can be applied to AP1000. Therefore, the means of resolution is applicable to AP1000.

The LOFTRAN IRWST model consisted of a single homogeneous fluid node and therefore did not account for the effects of stratification. Sensitivity studies showed that homogeneous treatment of IRWST fluid temperature produced conservative results for non-LOCA transients. The same treatment can be applied to AP1000. Therefore, the means for resolution is applicable to AP1000.

Assessments indicate that the AP1000 passive safety systems operate the same as the AP600, and that large margins to the regulatory limits exist for the transient events analyzed. It is expected that large margins will exist for the final accident analysis events analyzed with LOFTRAN.

2. The NOTRUMP code that was approved for AP600 can be used for the purposes of performing conservative (Appendix K) analysis of the small break LOCA events presented in Chapter 15 for AP1000. The basis for this conclusion is that for small break LOCA events, no new phenomenon is identified for AP1000 (when compared to AP600), and the test database that supported validation of this code for AP600 is applicable to AP1000. The means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000.

   It was noted in the AP1000 PIRT and Scaling Assessment (Reference 2) that some phenomena previously addressed for AP600 could be judged to be of higher importance for AP1000 (i.e., entrainment in the hot leg during the transition from ADS to IRWST injection of the SBLOCA event). To better address these phenomena, additional justification is provided as follows:

   - Sensitivity studies show that AP1000 SBLOCA performance is relatively insensitive to hot leg/upper plenum entrainment and that acceptable core cooling is maintained even when higher than expected entrainment (homogenous flow assumed in upper plenum, hot legs, and ADS-4) is assumed (Appendix F).

   - Comparison of the NOTRUMP level swell model to full scale bundle data confirms the validation of this aspect of NOTRUMP (Appendix G).

   - Comparison of NOTRUMP predictions to integral systems test data specific to AP1000 provide additional validation of NOTRUMP for AP1000 (Appendix E).

3. The WCOBRA/TRAC code that was approved for AP600 large break LOCA analysis can be used for the purposes of performing best-estimate analysis for AP1000. The basis for this conclusion is that for large break LOCA events, no new phenomena are identified for AP1000 (when compared to AP600) and the test database that supported validation of this code is applicable to AP1000. The means of resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000.

   The main code-related issue identified during the AP600 Design Certification review was the validation of WCOBRA/TRAC to address uniqueness of the passive safety system direct vessel injection (DVI). Westinghouse performed the validation and the NRC approved the code for AP600. As the AP1000 DVI configuration and location are the same as AP600, this validation is applicable to AP1000 as well. Therefore, the means of resolution applies to AP1000.

   The WCOBRA/TRAC computer code and large break LOCA methodology and approved by the NRC for AP600 are applicable to the 10CFR50.46 ECCS performance analysis of the AP1000 for 95th percentile calculated PCT values up to the 2200°F licensing limit.

4. The WCOBRA/TRAC code that was approved for AP600 long-term cooling analysis can be used for the purposes of performing conservative (Appendix K) analysis of long-term cooling for LOCA events presented in Chapter 15 for AP1000. The basis for this conclusion is that for LOCA events, no new phenomenon are identified for AP1000 (when compared to AP600), and the test database that supported validation of this code for AP600 is applicable to AP1000. The means for resolution of issues identified during the AP600 Design Certification review are applicable to the AP1000.

   The main code-related issues identified during the AP600 Design Certification Review included:

   - Application of WCOBRA/TRAC within the range of the OSU experimental validation, including the nodalization scheme used to perform the validation.

   - The use of "window" mode calculations of segments of the long-term cooling transient

   AP1000 scaling analysis demonstrates that the OSU test facility is sufficiently scaled to AP1000, so that the experimental validation is applicable to AP1000. Therefore the means of resolution applies to AP1000.

   In Reference 1, the use of WCOBRA/TRAC for long-term cooling in the "window" mode (as approved for AP600) was compared to analysis using a "continuous" mode for the limiting long-term cooling event. Results of that analysis demonstrated good agreement between the "window" mode analysis and the continuous mode analysis. Westinghouse has performed the limiting long-term cooling analysis using the continuous mode methodology presented in Reference 1, but retains the "windows" mode methodology for the less limiting events to minimize the resources expended to perform this analysis. Comparison of the results of the "continuous" mode to the "window" mode supports the assessment of conservative results for the "window" mode analyses. The means of resolution are therefore applicable to AP1000 and enhanced expanded use of continuous mode analysis.

The AP1000 passive safety systems provide large margins to the regulatory limits for accident analysis events analyzed with WCOBRA/TRAC for long-term cooling.

5. The WGOTHIC code that was approved for AP600 can be used for the purposes of performing conservative containment analysis of the events presented in Chapter 6 for AP1000. The basis for this conclusion is supported by the results of the AP1000 PIRT and Scaling assessment (Reference 2) and the assessment provided in this report that the means for resolution of code-related issues identified during the AP600 Design Certification review are applicable to AP1000.

The PIRT assessment found that for events that challenge containment integrity (i.e., large LOCA and large steam line break), no new phenomena are identified for AP1000 (when compared to AP600). The scaling assessment demonstrated that the range of important phenomena for AP1000 containment heat and mass transfer and liquid film coverage are sufficiently covered by the AP600 test database. Therefore, the extensive validation performed for use of the WGOTHIC code for AP600 is applicable to AP1000.

The main code-related issues identified during the AP600 Design Certification review included:

- Modeling circulation and mixing within containment
- Modeling PCS condensation and evaporation heat removal
- Validation of WGOTHIC evaluation model

The means of resolution of issues associated with modeling circulation and mixing within containment included applying results from experimental test facilities such as LST and HDR which showed that mixing circulation and mixing where enhanced when water is applied externally to the containment shell. In addition, conservative analysis code treatments such as eliminating heat transfer to floors and terminating heat transfer to conductors within dead-ended compartments after the blowdown phase is complete. Applying the LST and HDR tests for purposes of circulation and mixing behavior are as valid to AP1000 and they were to AP600 as the PCS design is the same. This is further confirmed by the CFD analysis presented in the AP1000 PIRT and Scaling Assessment. The conservative treatments are used for AP1000. Therefore, these means of resolution are still valid for AP1000.

The means of resolution of issues associated with modeling PCS condensation and evaporation heat removal included use of correlations with conservatively biased multipliers validated against separate effect heat and mass transfer tests. Initial PCS water coverage fraction was established from full-scale containment dome test data. The time dependent PCS water flow applied to the containment shell was the smaller of the PCS flow rate obtained assuming as single failure of one of two valves to open, or the estimated transient evaporation rate. Scaling analysis showed that the heat and mass transfer correlation ranges cover the range for AP1000 and the PCS dome test facility is fully applicable to AP1000. The conservative treatment of PCS flow rate will be used for AP1000. Therefore, these means of resolution are still valid for AP1000.

The means of resolution associated with validation of the WGOTHIC evaluation model included comparison against LST data. The scaling of the LST limited this comparison to the quasi-steady state portion of the transient as insufficient steam input distorted the rapid blowdown portion of

the transient. However, it was determined that the blowdown phase was not different than conventional plants for which there was ample validation for WGOTHIC. Therefore, the quasi-steady long-term cooling phase which relies on the passive safety features was well represented for AP600 and AP1000. Therefore, the means of resolution is still valid for AP1000.

The AP1000 has sufficient margin to the containment design pressure when bounding-type analyses are performed using WGOTHIC.

The following overall conclusions are reached supporting the applicability of the analysis codes to AP1000:

- The analysis codes were reviewed and approved by the NRC as part of the AP600 Design Certification process. The in-depth review conducted by the NRC staff included key elements of Draft Regulatory Guide DG-1096.

- PIRT assessment confirms that while there are a few phenomena that have been re-ranked, there are no new phenomena associated with the AP1000. Therefore, there are no models or features that must be added to the analysis codes and reviewed to account for any new phenomena.

- Scaling demonstrates that elements of the AP600 test database needed to validate the analysis codes for AP600 are applicable to AP1000. Therefore, as the extensive AP600 test program and code validation is applicable, the analysis codes do not need to be re-validated for AP1000.

- Analysis and evaluation of key plant parameters and accidents indicate that similar plant safety margins exist between AP600 and AP1000 and that AP600 and AP1000 behave similarly. Where margins were used in evaluating the acceptability of the AP600 safety analysis, sufficient margins have been established for the AP1000.

References

1. NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998.

2. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

APPENDIX A

DISCRETIONARY AND NON-DISCRETIONARY CHANGES MADE TO WCOBRA/TRAC-AP

The AP600 large break LOCA DCD analysis reports a calculated peak cladding temperature (PCT) at the 95th percentile of 1676°F, which occurs during the blowdown phase. In the 10CFR50.46 model assessments for 1998 an increase of 11°F was allocated to the AP600 PCT value. This permanent margin allocation considered the impact of the WCOBRA/TRAC Vessel Channel DX error. This, the licensing basis PCT for AP600 was re-established as 1687°F.

Subsequent to this assessment, further discretionary and non-discretionary changes to the WCOBRA/TRAC computer code were reported by Westinghouse for 1999 and 2000 in reference A-1 and Reference A-2, respectively. A description of each change relevant to the WCOBRA/TRAC version and/or the analysis methodology used in the AP600 CDC analyses is presented on the following pages. The conclusion is that the 1999 and 2000 changes have a 0°F impact on the AP600 LBLOCA and LTC analyses, and they also have no impact on the applicability of WCOBRA/TRAC-AP to the AP1000 DCD LBLOCA and LTC analyses.

References

1. NSBU-NRC-00-5970, "1999 Annual Notification of Changes to the Westinghouse Small Break LOCA and Large Break LOCA ECCS Evaluation Models, Pursuant to 10CFR50.46," Sepp, Westinghouse to J. S. Wermiel, May 12, 2000.

2. LTR-NRC-01-6, "10CFR50.46 Annual Notification and Reporting for 2000," Sepp, Westinghouse to J. S. Wermiel, March 13, 2001.

INCONSISTENT GUIDANCE FOR HOTSPOT OUTPUTS IN BE LBLOCA METHODOLOGY

Background

The BE LBLOCA methodology described in WCAP-12945-P-A contains inconsistent guidance on the selection of HOTSPOT outputs to be used as inputs for the 95th percentile PCT calculation. As a result, the published material does not always reflect the intended definition of late reflood, resulting in misrepresentation of the second reflood PCT time, magnitude and elevation for some transients which have low or non-existent second reflood PCTs. This issue was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

The impact of the inconsistent guidance for selection of HOTSPOT outputs was evaluated on a plant specific basis for all plants currently licensed with BE LBLOCA Evaluation Model. Only second reflood PCTs are affected by this inconsistency. The AP600 LBLOCA analysis exhibits no second PCT during reflood and therefore is unaffected.

DECAY HEAT UNCERTAINTY ERROR IN MONTE CARLO CALCULATIONS

Background

It was determined that an error existed in the calculation of decay heat uncertainty in the Monte Carlo code used for calculation of the 95th percentile PCT for Best Estimate LBLOCA. This issue was determined to be a Non-Discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

Plant specific PCT calculations were performed to assess the impact of this error for all analyses using the affected EMs. The correction for the AP600 LBLOCA analysis is calculated to be 0°F.

WCOBRA/TRAC GAP INPUT ERROR IN SECY UPI/BELOCA EM ANALYSES

Background

A survey of current SECY UPI, Best Estimate LBLOCA analyses and LBLOCA test simulations utilizing WCOBRA/TRAC identified an error in the application of the affected evaluation models. The error was in the specification of horizontal channel connections (gaps), which should be from lower numbered to higher numbered channel. The survey showed that only a few analyses contained this error. This error was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Potentially Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

For the AP600 Best Estimate LBLOCA analyses, no errors were found.

For the Oregon State APEX facility no errors were found

The survey found no errors in the AP600 LTC analysis.

GEDM INTERFACE ERROR

Background

A discrepancy between the inputs for the neutronics model and the way the code used the inputs was discovered that impacted the calculated gamma redistribution factors. This issue was determined to be a Non-Discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

It was determined that the error only concerns the neutronic input, which is not used in the code uncertainty/bias calculations, but only in plant calculations. A typical value of error in terms of the relative power is 0.001% or less than 0.01°F in peak average fuel temperature. This is well within the steady state tolerance criteria, such that estimated impact of the effect of this error on all plant calculations is 0°F, including AP600.

DROP DIAMETER PLOT TAPE STORAGE ERROR

Background

It was discovered the droplet diameter variable stored in the plot file contained a wrong value. This issue was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

There is no impact on analysis results, since the drop diameter edit output is not used in the calculation of PCT. A work around is available for old versions of the code. The WCOBRA/TRAC-AP code version corrects this error, and there is no PCT impact as a result of this error.

CLADDING OXIDATION EDIT ERROR

Background

It was determined that the hot rod fuel clad oxidation printouts after the end of fuel rod edits were incorrect. This issue was determined to be a Non-Discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

There is no impact on analysis results, since the guidance for the oxidation calculation uses the data in the plot file, which are correct. The WCOBRA/TRAC-AP code version corrects this error, and there is no PCT impact as a result of this error.

OUTPUT EDIT ERROR FOR SI UNITS

Background

It was determined that the fuel rod and 1D component edits were incorrect if the SI output option is selected. This issue was determined to be a Non-Discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

There is no impact on analysis results, since the reported PCT was not affected by this error. Users of older code versions have been advised to use English units for all WCOBRA/TRAC calculations. The current code version corrects this error. There is no PCT impact as a result of this error.

RADIATION HEAT TRANSFER TO VAPOR PHASE ERROR

Background

It was determined that the radiation heat transfer was set to zero when the void fraction in a channel exceeded 0.9999. This issue was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

Evaluations indicate that the single phase vapor heat transfer regime can occur during blowdown heatup, refill, and reflood. This error has negligible impact on existing analyses during the blowdown heatup and refill phases, since the single phase vapor heat transfer mode occurs only briefly in the blowdown heatup and refill. In reflood, single phase vapor conditions occur primarily during the downcomer boiling period for plants with late reflood PCTs. Under those conditions, the radiation heat transfer can account for approximately 20% of the total clad-to-vapor heat transfer. However, these conditions are nearly adiabatic, and the effect can be considered negligible for AP600, where the PCT occurs during blowdown. The WCOBRA/TRAC-AP code version corrects this error, and there is no PCT impact as a result of this error.

GRID HEAT TRANSFER ERROR

Background

It was determined that the grid's turbulence enhancement to heat transfer coefficient is erroneously applied to Radiation Heat Transfer to vapor phase. The enhancement from these grids should only be applied to the convective single phase heat transfer coefficient. This issue was determined to be a Non-Discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

The heat transfer multipliers used in the BE LBLOCA process include data from rod bundles with grids. Therefore, the effect of the error is compensated for by the multipliers, resulting in no impact on the analysis. The WCOBRA/TRAC-AP code version corrects this error, and there is no PCT impact as a result of this error.

PRESSURE DROP ERROR FOR 1D CONNECTIONS TO 3D VESSEL

Background

It was determined that the pressure drop was overestimated in the vertical momentum cell when the vessel vertical momentum flux is convected by the 1D component velocity. This issue was determined to be a non-discretionary change in accordance with Section 4.1.2 of WCAP-13451.

Affected Evaluation Models

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

In the AP600 large break LOCA simulation, the DVI ID component is connected to the downcomer vessel channel with a vertical connection. The impact of the pressure drop overestimation has been investigated and shown to be negligible on the AP600 large break LOCA transient.

During the AP600 long-term cooling transient fluid velocities are low in the vessel channels, so the pressure drop overestimation is negligible and does not impact the results predicted by the code.

There is no PCT impact on AP600 as a result of this error, which is corrected in the WCOBRA/TRAC-AP code version.

PAD 4.0 IMPLEMENTATION

Background

The Westinghouse Performance Analysis and Design Model (PAD) is used to generate fuel-related input data for use in LOCA licensing calculations. As documented in Reference 1, the Safety Evaluation Report for Version 4.0 of the PAD model was issued by the US NRC on April 24, 2000. Use of PAD Version 4.0 is considered to represent a Discretionary Change and will be implemented on a forward-fit basis, in accordance with Section 4.1.1 of WCAP-13451.

Affected Evaluation Models

1981 Westinghouse Large Break LOCA Evaluation Model

1981 Westinghouse Large Break LOCA Evaluation Model with BART

1981 Westinghouse Large Break LOCA Evaluation Model with BASH

1985 Westinghouse Small Break LOCA Evaluation Model with NOTRUMP

SECY UPI WCOBRA/TRAC Large Break LOCA Evaluation Model

1996 Westinghouse Best Estimate Large Break LOCA Evaluation Model

1999 Westinghouse Best Estimate Large Break LOCA Evaluation Model, Application to PWRs with Upper Plenum Injection

Estimated Effect

The implementation of PAD Version 4.0 with respect to Appendix K Large Break LOCA and Small Break LOCA analyses will be handled on a forward-fit basis and is assigned a PCT estimate of 0°F for 10CFR50.46 reporting purposes.

References

1. WCAP-15063-P-A Revision 1, with Errata, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)", J. P. Foster and S. Sidener, July 2000.

APPENDIX B

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APPENDIX C

ASSESSMENT OF NOTRUMP CODE ERRORS ON AP600 DSER ANALYSIS RESULTS

To provide evidence that the errors discovered in the AP600 NOTRUMP code do not invalidate the AP600 DSER results or the applicability of the revised code to the AP600/AP1000 designs, information for AP600 2-Inch Cold Leg break simulations was generated. A synopsis of the results is provided below. Note that NOTRUMP Version 36.0, which was released after the AP600 SSAR analysis was performed, contains discretionary changes applicable to all Westinghouse plant designs as well as the gravitational head error correction.

The assessment of the impact of errors associated with implicit fluid node gravitational head, droplet fall models and volumetric flow link variable updating can be shown to have a negligible impact on the AP600 2-Inch Cold Leg break when compared to the AP600 DSER results. Figures C-1 through C-3 present comparisons of the Pressurizer pressure (Figure C-1), Core/Upper plenum mixture level (Figure C-2) and RCS system inventory (Figure C-3) responses associated with the correction of the fluid node gravitational head in Version 36.0 and the droplet fall model errors respectively. Figures C-4 through C-6 present the same figures associated with the responses when the volumetric flow link variable updating error is corrected in NOTRUMP Version 37.0. As can be seen by reviewing these figures and the sequence of events summary in Table C-1, the conclusion that the impact of these errors on the code and the simulation results is negligible can be readily supported.

The errors associated with the region depletion model logic can not be directly assessed since AP600 plant specific transient simulations have not been performed with a corrected code version. As a result, only the impact established from traditional PWR designs can be utilized to make this determination. The documentation supporting correction of this error contains the following synopsis.

"Although this is a code correction, the impact is expected to be minimal since the interior metal node temperature updates performed in the old code version were only out-of-phase by one time step with respect to the interior fluid node central variable adjustments. In addition, the interior fluid node central variable adjustments are expected to be small and to occur infrequently during a typical transient. As such it is expected that the internal metal node temperatures are considered to have an insignificant impact on analysis results, since the temperature differences between un-heated conductors and adjacent fluid channels are typically small."

To further substantiate this conclusion, plot results for three key parameters (Core mixture level, core exit vapor temperature and core exit vapor flow) were generated for several Westinghouse plant cases. The results of these cases demonstrate the benign nature of this code error correction and support the conclusion that small break LOCA transient simulations are negligibly impacted by this change. In addition, since the AP600/AP1000 plant results obtained do not indicate the existence of core uncovery, it is expected that the AP600/AP1000 designs will also exhibit no impact from this change.

Table C-1 Sequence of Events Summary

Event	SSAR Results (Seconds)	Version 36.0 Results (Seconds)	Revised Droplet Fall Results (Seconds)	Version 37.0 Results (Seconds)
Break Opens	0.0	0.0	0.0	0.0
Reactor Trip Signal	33.5	33.5	33.5	33.5
"S" Signal	39.7	39.7	39.7	39.7
ADS Stage 1	1032	1036	1036	1036
ADS Stage 2	1102	1106	1106	1106
ADS Stage 3	1222	1226	1226	1226
Accumulators Empty	1470	1468	1467	1467
ADS Stage 4	2422	2414	2418	2401
Core Makeup Tank Empty	2820	2790	2790	2772
IRWST Injection Starts	3544	3548	3561	3562

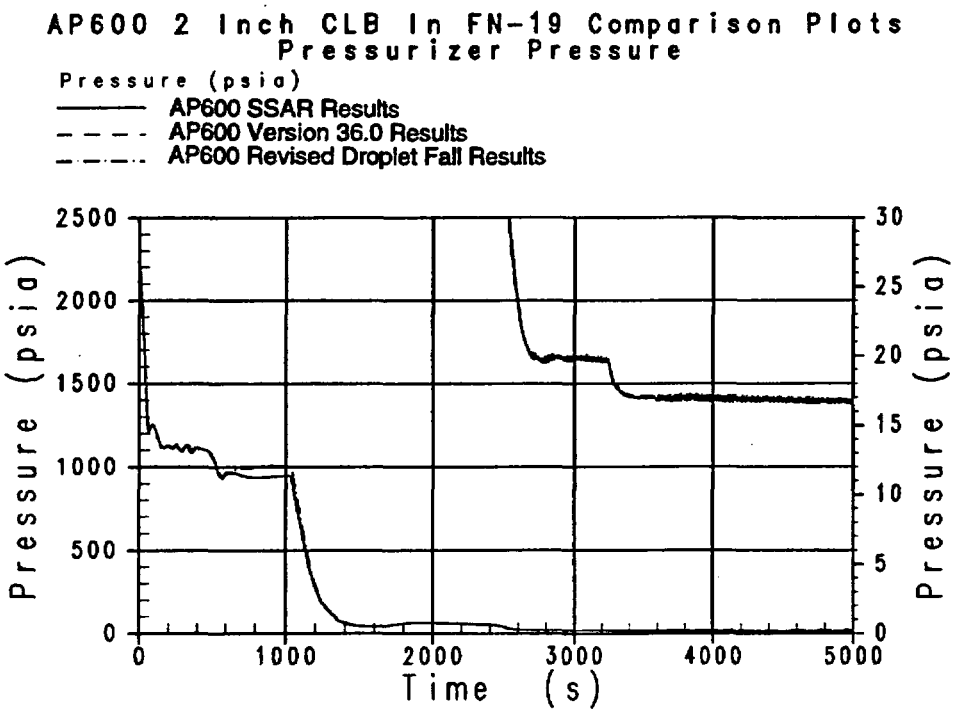

Figure C-1 Pressurizer Pressure

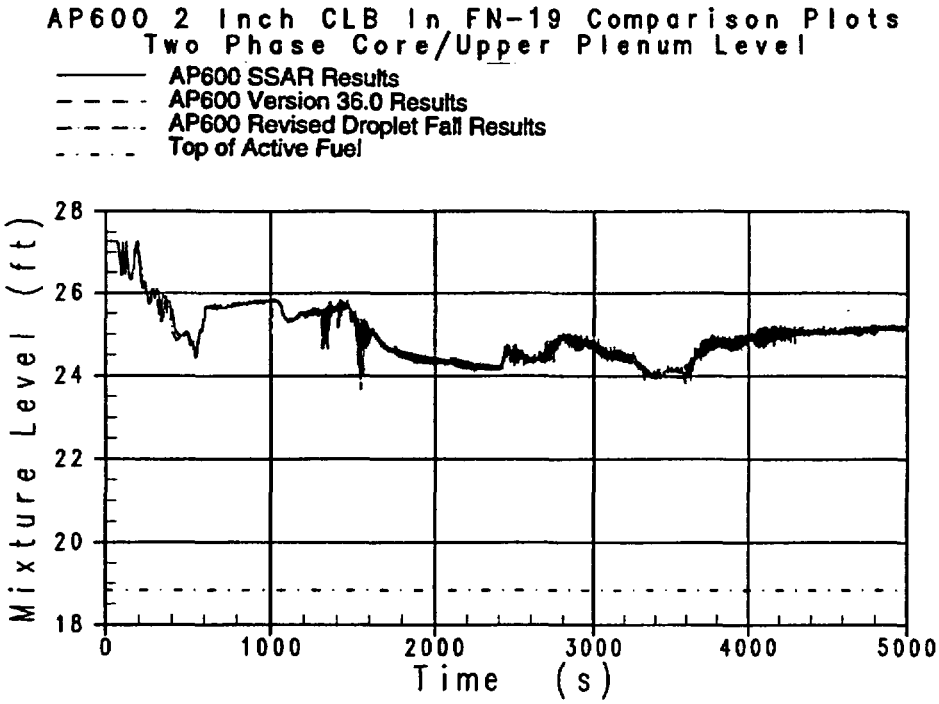

Figure C-2 Core/Upper Plenum Mixture Level

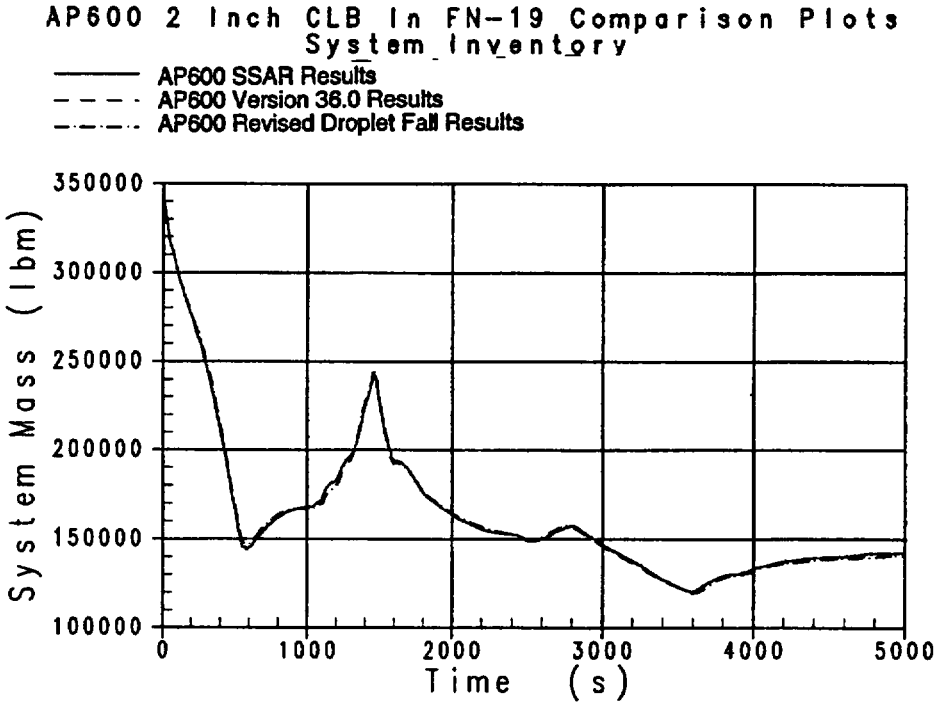

Figure C-3 RCS System Inventory

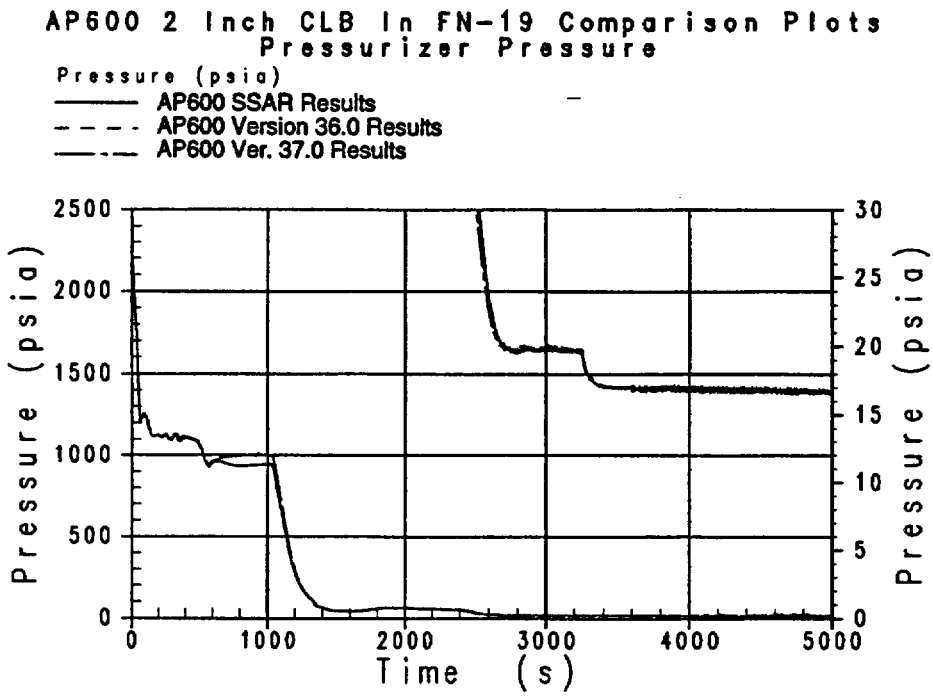

Figure C-4 Pressurizer Pressure

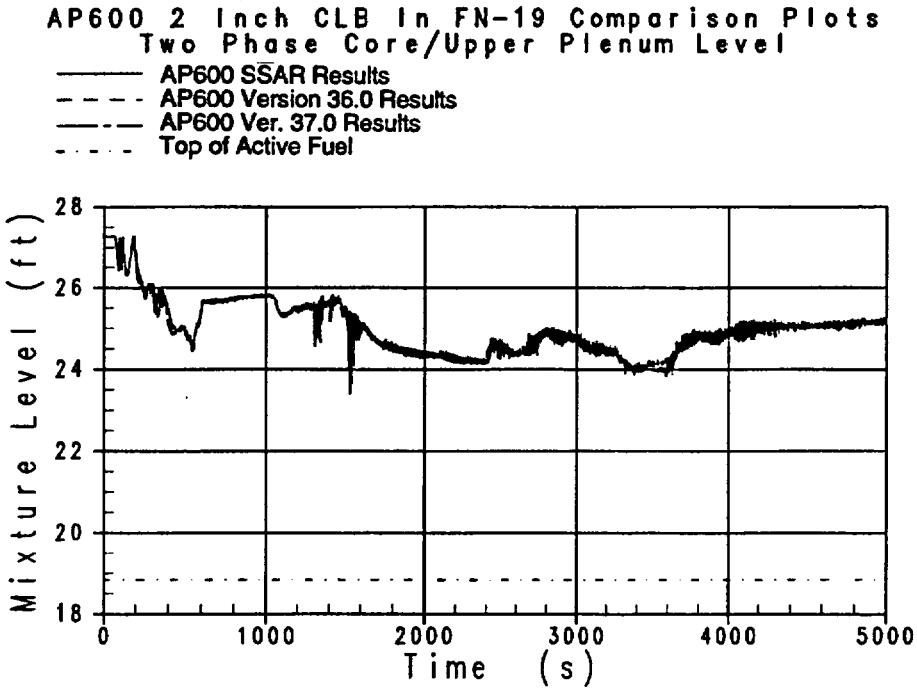

Figure C-5 Core/Upper Plenum Mixture Level

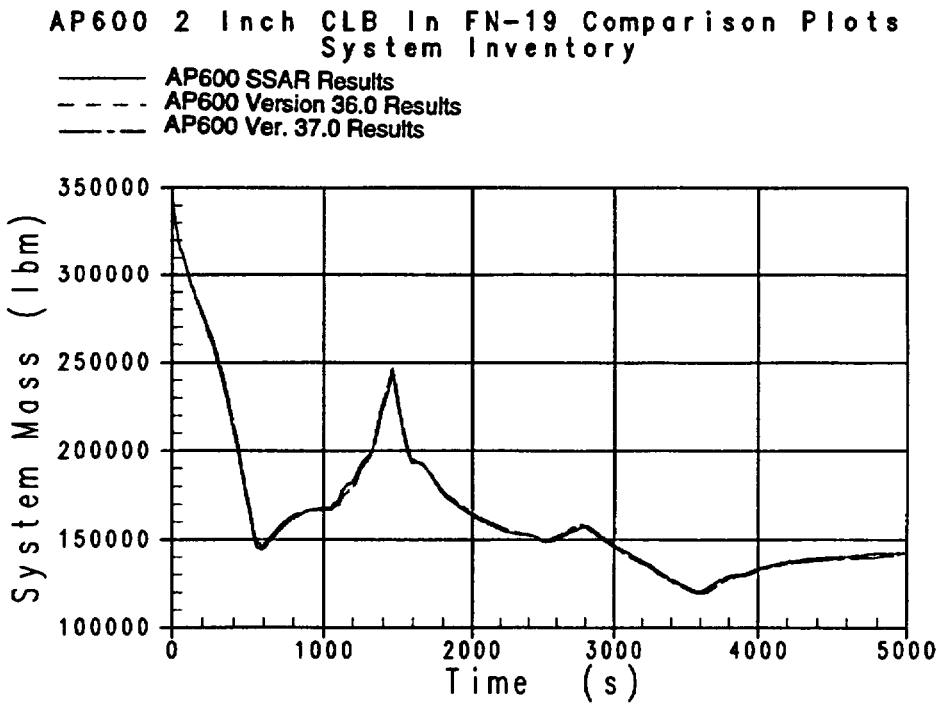

Figure C-6 RCS System Inventory

APPENDIX D

EVALUATION OF PRHR HEAT EXCHANGER TUBE EXTERNAL HEAT TRANSFER

The PRHR heat exchanger is a C-tube design with 689 tubes. The heat exchanger is located in the in-containment refueling water storage tank (IRWST), and serves as the safety-grade decay heat removal mechanism for design basis accidents. The heat exchanger is normally isolated from the reactor coolant system. In the event of an "S" signal, the isolation valves are opened and RCS water enters the heat exchanger from the hot leg. Cold water is returned to the cold leg at the reactor coolant pump suction. Natural circulation flow is generated in the heat exchanger by the density difference between the hot inlet flow and the cold outlet flow and the separation between the thermal center of the heat exchanger and the core.

D1.0 PRHR HEAT EXCHANGER TUBE HEAT TRANSFER MODEL

At any point along the length of the PRHR tube, the resistance to heat transfer from the fluid inside the tubes to the IRWST water outside the tubes is comprised of three components: the film drop inside the tubes, the thermal conductivity of the tube wall, and the film drop outside the tubes.

$$q = (Tin - T\infty) / (R1 + R2 + R3) \quad (1)$$

where Tin is the temperature of the fluid inside the tubes
$T\infty$ is the local bulk temperature in the pool outside the tubes

and R1, R2, and R3 are the three resistances described above

$$R1 = 1 / (h_{in} * \pi * Di * \Delta L) \quad (2)$$

$$R2 = \ln (Do / Di) / (2\pi * ktube * \Delta L) \quad (3)$$

$$R3 = 1 / (h_{out} * \pi * Do * \Delta L) \quad (4)$$

where h_{in} is the heat transfer coefficient inside the tubes
h_{out} is the heat transfer coefficient outside the tubes
Do is the outside tube diameter
Di is the inside tube diameter
k_{tube} is the thermal conductivity of the tube wall

and ΔL is the differential length of the tube segment

D1.1 HEAT TRANSFER INSIDE PRHR TUBES

The inlet flow to the PRHR heat exchanger can be either single-phase liquid or two-phase mixture. For single-phase liquid inside the tubes, the heat transfer coefficient is described by the Dittus-Boelter correlation:

$$h_{db} = 0.023 * Re^{0.8} * Pr^{0.4} \quad (5)$$

where Pr is the Prandtl number of the fluid inside the tube

and Re is the Reynolds number given by

$$Re = 4 * m / (\pi * Di * \mu) \quad (6)$$

where m is the flow rate of the fluid in the tube

and μ is the dynamic viscosity of the fluid in the tube

For two-phase mixture, the Shah condensation model is used (Ref. D3).

$$h_{shah} = h_{db} * (1 - x)^{0.8} + 3.8 * x^{0.04} / (p / 3208)^{0.38} \quad (7)$$

where p is the saturation pressure inside the tube

and x is the flow quality

Thus,

$$h_{in} = \begin{cases} h_{db} & \text{for } x = 0 \\ h_{shah} & \text{for } x > 0 \end{cases} \quad (8)$$

D1.2 HEAT TRANSFER OUTSIDE PRHR TUBES

An extensive test program was conducted to provide heat transfer characteristics for the PRHR heat exchanger (Ref. D1). The results of these tests showed that the heat transfer from the outside of the tubes is characterized by either free convection or nucleate boiling depending on the outer wall temperature of the tubes and the local pool conditions. Free convection is described by McAdams' correlation:

$$h_c = 0.13 * k / L * [Gr * Pr]^{1/3} \quad (9)$$

where k is the water thermal conductivity
L is the characteristic dimension
Pr is the Prandtl number of the fluid outside the tube

and Gr is the Grashof number which is given by

$$Gr = g * \beta * (Tw - T\infty) * L^3 / \nu^2 \quad (10)$$

where g is the gravitational constant
β is the liquid volumetric expansion coefficient
Tw is the outer tube wall temperature
$T\infty$ is the bulk temperature in the pool

and ν is the liquid kinematic viscosity

Combining equations 9 and 10,

$$h_c = 0.13 * k * [g * \beta * (Tw - T\infty) * Pr / \nu^2]^{1/3} \quad (11)$$

For the case where the tube outer wall temperature is greater than the local saturation temperature in the pool, the water will boil. Reference D1 showed that the boiling heat transfer was degraded somewhat at the top of the tube bundle as steam generated further down blanketed the upper portions of the tubes. A correlation was generated from the test data based on the Rosenhow correlation and used in LOFTRAN (Ref. D2):

$$q/A = \mu_f * h_{fg} * [\, g*(\rho_f - \rho_g)/(g_c*\sigma)\,]^{0.5} * [\, c_p * \Delta T / (C_{sf}*Pr*h_{fg}\,]^{1/0.4523} \quad (12)$$

where q/A is the heat flux
μ_f is the liquid dynamic viscosity
h_{fg} is the heat of vaporization
g is the acceleration due to gravity
g_c is the gravitational constant
ρ_f is the liquid density
ρ_g is the vapor density
c_p is the liquid specific heat
σ is the liquid surface tension
Pr is the liquid Prandtl number
C_{sf} is a constant derived from the test data = 0.0413

and ΔT is the temperature difference Tw - Tsat

Equation 12 can be written as

$$q/A = h_{nb\text{-}loft} * (\, Tw - Tsat\,) \quad (13)$$

where $h_{nb\text{-}loft}$ is the nucleate boiling heat transfer coefficient used in LOFTRAN.

$$h_{nb\text{-}loft} = a * (\, Tw - Tsat\,)^b \quad (14)$$

where the constant a is dependent on the local pool conditions

$$a = \mu_f * h_{fg} * [\, g*(\rho_f - \rho_g)/(g_c*\sigma)\,]^{0.5} * [\, c_p / (C_{sf}*Pr*h_{fg}\,]^{1/0.4523} \quad (15)$$

and the constant b is given by

$$b = 1 / 0.4523 - 1 = 1.2109 \quad (16)$$

The NOTRUMP code uses a global nucleate boiling model for all heat transfer surfaces and does not allow differentiation between the PRHR tubes and other surfaces such as the fuel rods. The code uses the Thom correlation (Ref. D3):

$$h_{nb\text{-}not} = (\, 0.072\,)^{-2} * e^{(P/630)} * (\, Twall - Tsat\,) \quad (17)$$

where P is the local pressure in the pool

Either equation 14 or 17 can be used to calculate the nucleate boiling heat transfer coefficient if the tube outer wall temperature is greater than the local saturation temperature. The nucleate boiling coefficient is compared to the natural circulation coefficient from equation 11 and the maximum is used.

$$h_{out} = MAX\ (\ h_c\ ,\ h_{nb}\ ) \tag{18}$$

This is the value used in Equation 4 to calculate the resistance to heat transfer outside the tube.

D1.3 OVERALL HEAT TRANSFER IN TUBES

After the overall heat transfer is calculated for a tube segment using Equation 1, the outside wall temperature is calculated by

$$Tw = Tin - q * (\ R1 + R2\ ) \tag{19}$$

The process is repeated until the heat transfer, q, converges, and the solution for the tube segment has been determined depending on whether the flow is single-phase or two-phase. For single-phase flow, the temperature of the fluid inside the tube exiting this segment is lower due to this heat transfer.

$$Tin_{i+1} = Tin_i - q\ /\ (\ m * c_p\ ) \tag{20}$$

where Tin_{i+1} is the fluid temperature for the next segment
Tin_i is the fluid temperature of the previous segment

and c_p is the specific heat of the fluid inside the tube

For two-phase flow, the enthalpy change is given by

$$h_i{+}1 = h_i - q\ /\ m \tag{21}$$

where h_{i+1} is the fluid enthalpy for the next segment

and h_i is the fluid enthalpy for the previous segment

The quality for the next segment is given by

$$x_{\ i+1} = (\ h_{i+1} - h_f\ )\ /\ (\ h_g - h_f) \tag{22}$$

where h_f is the saturated liquid enthalpy at the pressure inside the tube

and h_g is the saturated vapor enthalpy at the pressure inside the tube

If the enthalpy for the next segment is less than or equal to the saturated liquid enthalpy, the flow is assumed to be single-phase liquid.

The process is repeated for all segments of the tube.

The overall heat transfer from the PRHR is calculated by summing the individual segments over all of the tubes

$$Q_{tot} = [\Sigma q_i] * N_{tubes} \quad (23)$$

where N_{tube} is the total number of tubes in the heat exchanger

D2.0 DETERMINING EFFECT OF NUCLEATE BOILING CORRELATION

D2.1 SINGLE-PHASE EFFECT

Several calculations were made to determine the effect of the nucleate boiling correlation. A typical PRHR flow rate of 500,000 lbm/hr is assumed along with an inlet temperature of 300°F. The inlet flow is assumed to be single-phase liquid. This corresponds to 0.2 lbm/s per tube and 1.52 ft/s velocity.

The heat transfer calculation described in the previous section was performed using both the NOTRUMP and LOFTRAN nucleate boiling correlations. In both cases, the tops of the tubes experience boiling and transition to natural convection as the fluid temperature inside the tubes decreases and the pool water pressure increases along the vertical portion of the tubes. The heat transfer coefficient as a function of length along the tubes is shown in Figure D-1. This figure shows that the Thom correlation predicts significantly higher film coefficients for nucleate boiling than the modified Rosenhow correlation.

Figure D-2 shows the local heat transfer rate as a function of length along the tubes. This plot shows that although the film coefficient in the boiling region is higher, the wall temperature is lower and the heat transfer rates are only moderately higher. In addition, the higher heat removal in the beginning of the tubes results in lower fluid temperatures inside the tubes in the lower region as is shown in Figure D-3. Thus, more heat is removed in the lower region for the case where the modified Rosenhow correlation is used. The overall heat removal for the heat exchanger was 11.9 MW for the Thom case and 11.2 MW for the modified Rosenhow case. Thus, the current NOTRUMP model overpredicts the PRHR heat transfer by about 6 percent for these typical conditions.

Reference D3 recommends a reduction in the PRHR heat transfer area of 50 percent when the fluid velocity inside the tubes exceeds 1.5 ft/s. A separate calculation was performed to determine the effect of using the Thom correlation with a 50 percent reduction in the heat transfer area. The resulting heat removal for the heat exchanger is 9.8 MW, which is a reduction of about 13 percent from the modified Rosenhow case. Thus, it is conservative to reduce the PRHR heat exchanger area by 50 percent to account for the use of the Thom correlation.

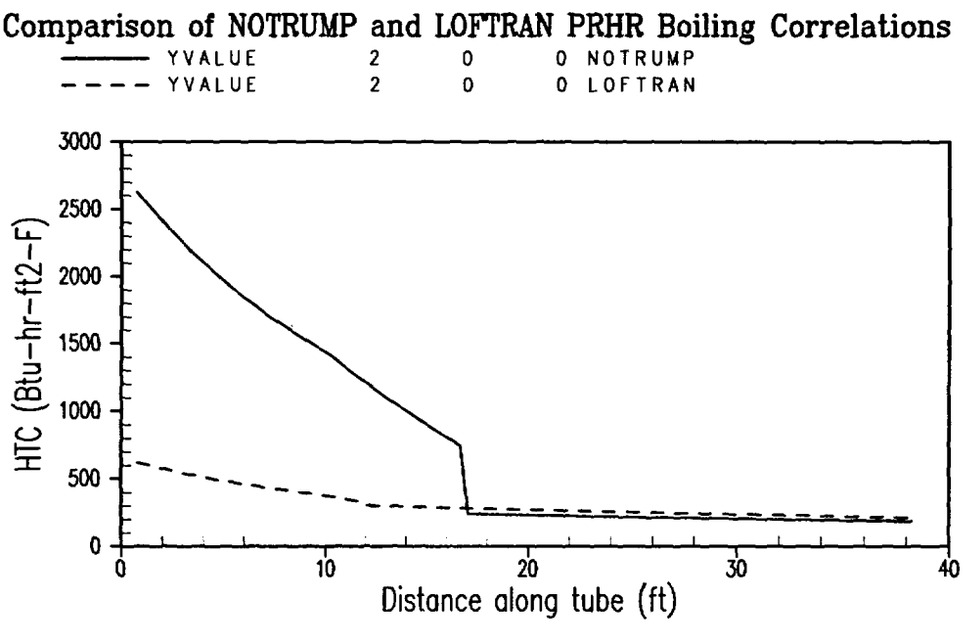

Figure D-1 Heat Transfer Coefficients Versus Distance Along Tube – Single Phase

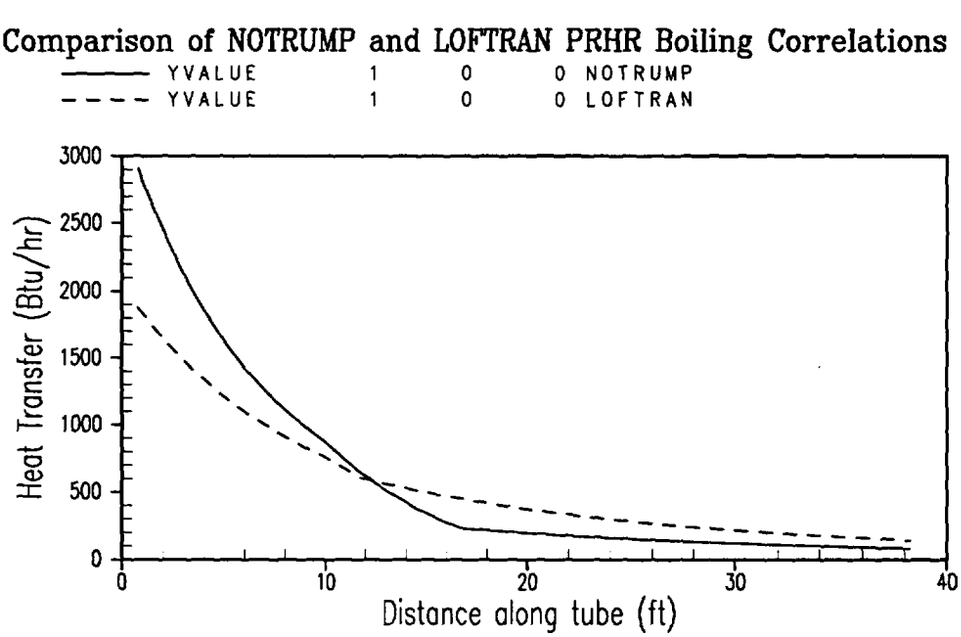

Figure D-2 Heat Transfer Rate Versus Distance Along Tube – Single Phase

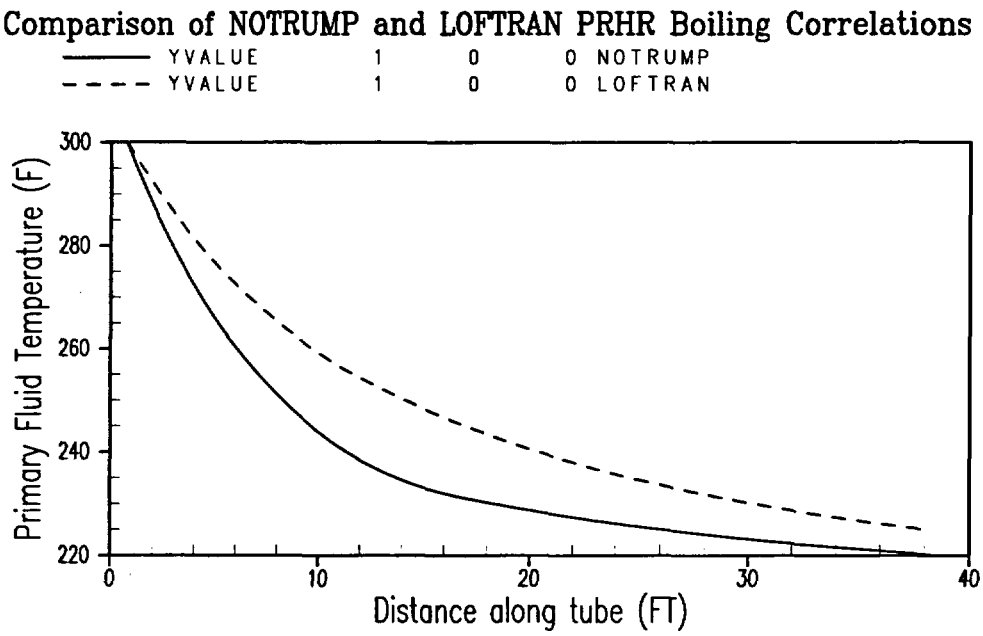

Figure D-3 Primary Fluid Temperature Versus Distance Along Tube – Single Phase

D2.2 TWO-PHASE EFFECT

For the case of two-phase mixture entering the PRHR heat exchanger, the heat transfer is higher in the condensing region. For this case, the same conditions are assumed: 500,000 lb/hr inlet flow at 300°F with an IRWST temperature of 212°F. However, for this case, the inlet flow is assumed to be two-phase with a flow quality of 0.05. As before, two cases are analyzed: one using the Thom correlation for boiling on the outside of the tubes, and one using the modified Rosenhow correlation.

Figure D-4 shows that the heat transfer coefficient is higher for a larger portion of the tube length using the Thom correlation. Figure D-5 shows that there are significantly higher heat transfer rates when the tubes are condensing two-phase mixture for the case using the Thom correlation. However, the vapor is condensed within a shorter tube length for this case, and in the natural convection region the higher fluid temperature inside the tubes results in higher heat transfer for the case where the modified Rosenhow correlation is used. This result is also shown in Figure D-6 where the fluid temperature remains at the inlet temperature until the vapor is condensed, then falls more rapidly using the Thom correlation.

Using the Thom correlation, the overall heat removal was 17.3 MW, as compared with 16.3 MW using the modified Rosenhow correlation (~6 percent increase). An additional run was made using the Thom correlation and reducing the tube heat transfer area by 50 percent. The overall heat removal for this case is 14.5 MW which is approximately 11 percent lower than the modified Rosenhow correlation. Thus, for two-phase flow into the PRHR heat exchanger, the Thom correlation with a 50-percent decrease in the PRHR heat transfer area conservatively underpredicts the PRHR heat transfer when compared to the modified Rosenhow correlation.

D3.0 CONCLUSIONS

The results of this study show that the use of NOTRUMP with the Thom nucleate boiling correlation slightly overpredicts the heat removal by the PRHR heat exchanger for both single-phase and two-phase inlet flow. By reducing the heat transfer area by 50 percent, the heat removal rate is conservatively underpredicted by the correlations in NOTRUMP by 11 to 13 percent when compared to the modified Rosenhow correlation used in LOFTRAN.

D4.0 REFERENCES

D1. WCAP-12980, Revision 3, AP600 PRHR Heat Exchanger Final Test Report, April 1997.

D2. WCAP-14234, Revision 1, LOFTRAN and LOFTTR2 AP600 Code Applicability Document, August 1997.

D3. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, August 1998.

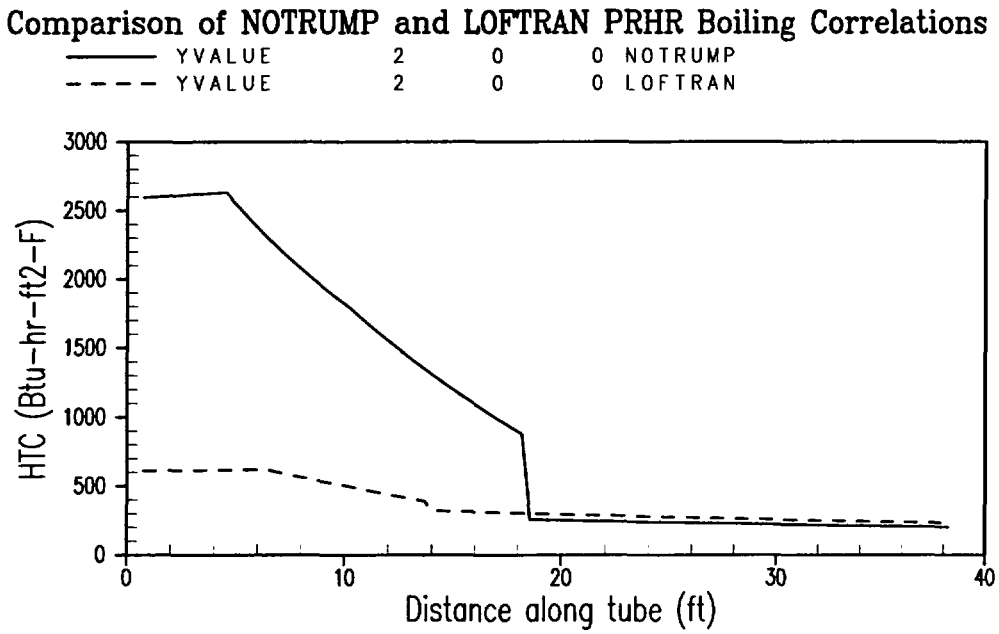

Figure D-4 Heat Transfer Coefficients Versus Distance Along Tube – Two Phase

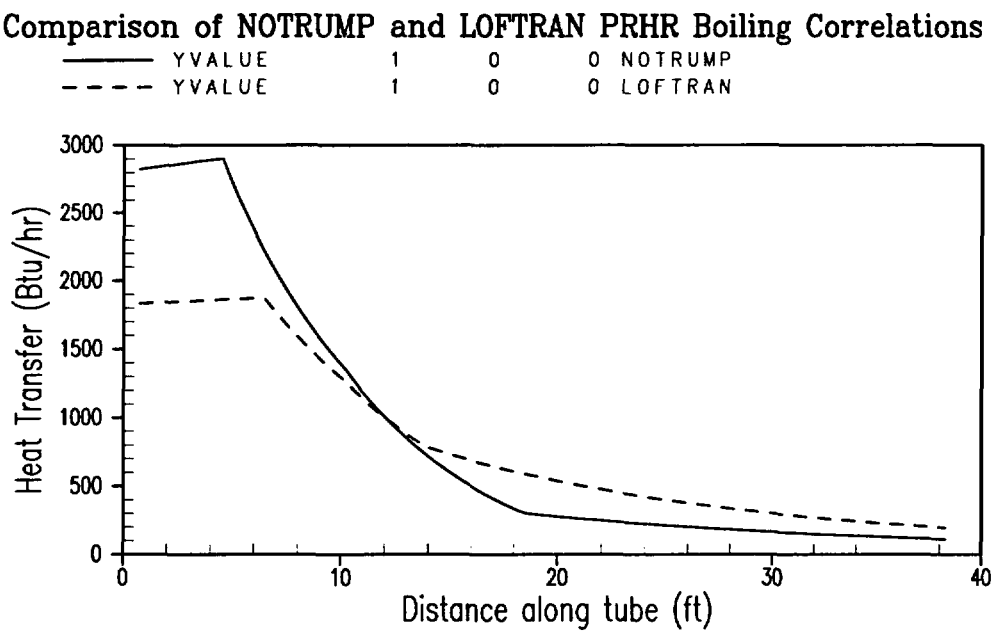

Figure D-5 Heat Transfer Rate Versus Distance Along Tube – Two Phase

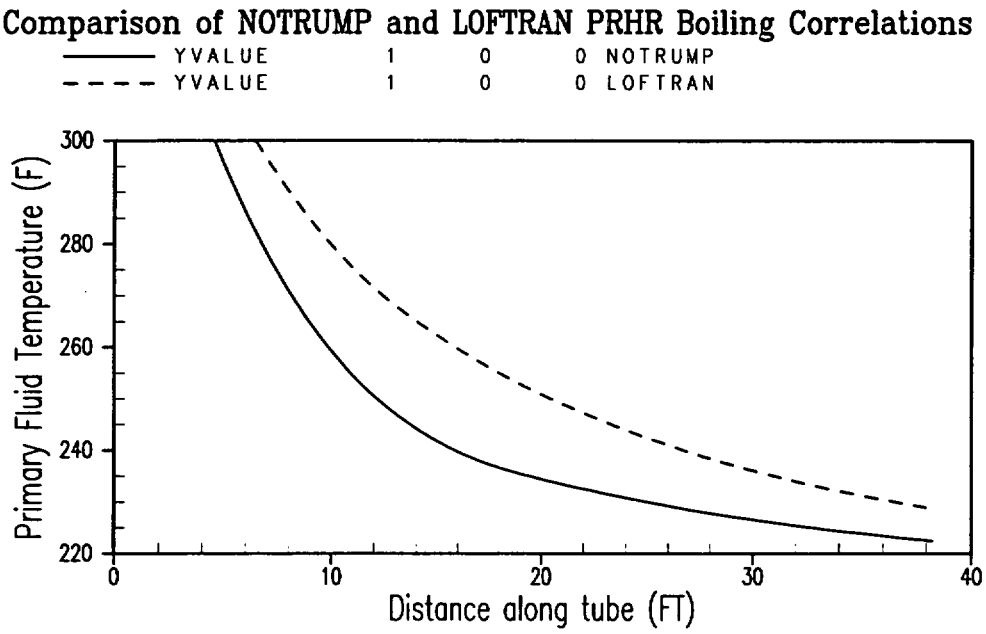

Figure D-6 Primary Fluid Temperature Versus Distance Along Tube – Two Phase

APPENDIX E

SIMULATION OF APEX-1000 TEST FACILITY WITH NOTRUMP – AP600

E1.0 BACKGROUND

To further confirm the applicability of the NOTRUMP computer code to predict the AP1000 plant behavior for small break loss of coolant accidents (SBLOCAs), the revised OSU APEX test facility (References E1 and E1) was modeled with the Advanced Plant version of the NOTRUMP computer code. The noding diagram used for the Reference E1 OSU APEX simulations is found in Figure E-0. The model used for these simulations is similar to that used for the AP600 APEX simulations with the following exceptions:

- Revised noding in the pressurizer
- Revised noding in the core makeup tanks

The pressurizer noding was altered from a single fluid node to multiple fluid nodes (See Figure E-1) for several reasons. First, the APEX facility was modified to accommodate the increase in pressurizer volume required to represent the AP1000 plant design. The modification was such that a section was added to the upper pressurizer. This upper section is a larger diameter than the lower section. Therefore, to properly model the change in geometry requires an additional fluid node be added to the NOTRUMP model. In addition, to improve the predicted void distribution in the pressurizer, additional fluid nodes were added to represent the pressurizer surge line and split the common fluid node section, representing the pressurizer tank, into [ ]^{a,c} individual fluid nodes as can be seen in Figure E-1.

The core makeup tank (CMT) model was revised to add additional fluid nodes to enhance the fluid temperature distribution predicted by the NOTRUMP code. Since the NOTRUMP code does not have a thermal stratification model, when warm fluid is introduced to a fluid node, it is assumed to perfectly mix with the existing fluid node. As such, when only a few fluid nodes are modeled, the fluid temperature at the bottom of the CMT begins to artificially heat due to the numerical mixing effect. A sensitivity study was performed which altered the CMT noding from the standard [ ]^{a,c} model to a [ ]^{a,c} model in the AP600 APEX test series in RAI response 440.339 (Reference D1). The conclusions of this sensitivity study were as follows:

> *The conclusions of this study is that using more nodes in the CMTs represents a way to approximately simulate the CMT thermal stratification effects, to help account for the lack of a CMT thermal stratification model in NOTRUMP. This technique can be used to improve the CMT outlet temperature behavior in small break transients. This CMT noding study supports the conclusions of the independent assessments that are being conducted for the preparation of the summary section for Revision 2 of the NOTRUMP Final Validation Report for AP600. The summary section will indicate that the lack of a CMT thermal stratification model and the coarse noding used lead to significant differences in the CMT outlet temperature and resulting small break transient, but that the continued use of the [ ]^{a,c} CMT model is acceptable because its effect on the transient is conservative (high core void fraction, delayed ADS).*

The CMT noding used for the studies presented herein are shown in Figure E-2. For the transient results presented herein, the use of the increased CMT noding will not have a significant effect due to the time frame over which the CMTs are emptied for the DVI line break simulations. This was subsequently confirmed via the performance of a CMT noding sensitivity study for the APEX-1000 simulations where

the original Reference E1 nodalization was used. As expected, revising the CMT noding had little effect on the DEDVI transient simulation results.

The APEX model simulations differ from that used in the AP1000 plant design as well. The same differences described above also apply to the modeling differences between the plant model and the OSU model. However, as described above, the CMT noding differences result in a conservative prediction of ADS actuation times and core average void fraction predictions.

Two OSU APEX test simulation results were performed with the Advanced Plant version of the NOTRUMP computer code. These were:

- Test DBA-02, Double-Ended Direct Vessel Injection Line Break with an ADS-4 single failure on the pressurizer side (ADS 4-2).

- Test DBA-03, Double-Ended Direct Vessel Injection Line Break with an ADS-4 single failure on the non-pressurizer side (ADS 4-1).

The DEDVI line break represents the most severe accident for the AP1000 plant design in that it eliminates a full train of makeup capability. The modeling methodology used for the APEX simulations is the same as that used for the plant simulations with the following exceptions.

- No passive residual heat exchanger heat transfer [ ]^{a,c} was applied.

- The ADS-4 flow paths were modeled with the [ ]^{a,c} during the transition to noncritical conditions and, subsequently, the orifice equation for post-critical flow.

The methodology used to model the ADS-4 flow paths in the OSU simulations differed from that used for the AP1000 plant analysis. In the AP1000 plant simulations, the ADS-4 flow paths are altered from [ ]^{a,c} flow paths to [normal interior]^{a,c} flow links once noncritical conditions have been reached in both ADS-4 paths. At that time, the FLOAD4 resistance adjustment factor (Reference E1) of [ ]^{a,c} is placed on both ADS-4 flow paths and the transient simulation is continued. Note that the plant and OSU test facility differ in the ADS-4 flow path in that for the plant, the ADS-4 squib valve is the last component in the path and discharges directly to containment. For APEX, the squib valve is represented by a flow venturi with subsequent piping to the ADS-4 separator. This level of detail is not represented in the NOTRUMP model for the APEX test facility. For the APEX simulations performed herein, the ADS-4 flow links in the NOTRUMP model used the [ ]^{a,c}. This model was selected based on the results of comparisons of predicted with measured ADS-4 flow. In order to assess the effect of the change in modeling methodology on the AP1000 plant results, the DEDVI line break, assuming atmospheric containment conditions, was re-performed with the same modeling assumption as used for the APEX test facility [ ]^{a,c}. The results obtained indicate only a minor change in the predicted ADS-4 behavior and, subsequently, IRWST injection behavior. To further supplement this conclusion, the APEX-600 test series DEDVI line break (Test SB12) was also re-performed using the revised ADS-4 methodology. Again, only minor differences in ADS-4 flow and, subsequently, IRWST injection times were observed.

The NOTRUMP simulation result comparisons of tests DBA-02 and DBA-03 are presented below. The sequence of events for these tests is provided in Tables E1 and E2.

E1.1 COMPARISON OF NOTRUMP SIMULATION TO TEST DATA FOR TEST DBA-02

Figure E-3 and Figure E-4 compare the pressure at the top of the pressurizer and downcomer regions for the test and the NOTRUMP simulation. The pressure decreases initially due to the blowdown through the break. The depressurization rate slows (and stops for NOTRUMP) when the primary system becomes saturated. Following actuation of ADS-1 at [ ]^{a,b} seconds in the test (81.3 seconds for NOTRUMP), the depressurization rate increases significantly. The downcomer pressure is provided since this pressure ultimately controls the onset of intact IRWST (IRWST-2) injection. The trends observed in the downcomer pressure closely follow that observed in the pressurizer. The agreement between the test data and the prediction are reasonable since the trends observed are similar.

Figure E-5 shows the collapsed liquid level in the pressurizer for the test and the NOTRUMP simulation. The break flow causes a rapid decrease in pressurizer level and empties the pressurizer at approximately [ ]^{a,b} seconds for the test and 70 seconds for the NOTRUMP simulation. The pressurizer level increases following ADS actuation for both the test and the simulation with NOTRUMP initially refilling faster than the test until ADS-2 actuation. The NOTRUMP simulation collapsed mixture level recovers to slightly lower level following ADS actuation compared to that observed in the test facility. Following ADS-4 actuation, both the test and NOTRUMP simulations indicate a period of continued pressurizer refill until the ADS-4 flow paths become dominant. The collapsed level predicted by NOTRUMP decreases in a similar manner to that observed in the test following ADS-4 actuation. Therefore, the NOTRUMP results are considered to be in reasonable agreement with the test data.

Figure E-6 and Figure E-7 show the collapsed liquid levels in CMT-1 and CMT-2 for the test and the NOTRUMP simulation respectively. In the test, CMT-1, which is attached to the broken DVI line begins draining out the break at [ ]^{a,b} seconds in the test compared to 20 seconds for the NOTRUMP simulation. This can also be seen in the CMT injection flow plots (Figure E-8 and Figure E-9). As such, the NOTRUMP simulation transitions from re-circulation to draindown mode earlier than observed in the test and subsequently predicts higher injection flows. CMT-2 transitions from recirculation to draindown mode at about [ ]^{a,b} seconds (150 seconds for NOTRUMP). The comparisons indicate that the NOTRUMP intact CMT drains slightly earlier than observed in the test. This is due to the earlier predicted emptying of the intact accumulator (Figure E-15 and Figure E-27). The conclusions that can be reached are that the NOTRUMP CMT predictions are considered to be in reasonable agreement with the test data. The underprediction of the transition to CMT-1 draindown mode negligibly impacts the predicted ADS-1 actuation time compared to the test results and is considered reasonable.

Figure E-10 through Figure E-13 present the collapsed steam generator level comparisons between the test and NOTRUMP simulations. As can be seen, the NOTRUMP results are in good agreement with the test data.

Figure E-14 and Figure E-15 present the collapsed liquid levels in accumulator 1 and accumulator 2 for the test and the NOTRUMP simulation. The comparison between the test and the NOTRUMP simulation is considered good for both accumulators with the interruption of accumulator 1 discharge appropriately

presented by the NOTRUMP simulation following the transition of CMT-1 from recirculation to draindown mode.

The next series of plots relate to the collapsed and two-phase levels at different locations in the vessel. The trends of the simulation plots are in reasonable agreement with the test up to approximately ADS-2 actuation. Following ADS-2 actuation, the test and NOTRUMP simulations diverge as a result of the test-observed, two-dimensional downcomer behavior, which cannot be modeled with the NOTRUMP [ ]^{a,c} downcomer (See Reference E1 for additional details). A review of the core inlet temperature (Figure E-28) indicates that the NOTRUMP simulation is predicting sub-cooled conditions whereas the test indicates saturated core entry exist. This can be partly attributed to the lack of two-dimensional downcomer modeling and partly due to heating of the intact DVI injection flow as it impinges on the core barrel. The NOTRUMP model has appropriate heat transfer models from fluid to metal structures in the downcomer fluid node but does not account for the heating of the injected fluid as it impinges onto the core barrel. As such, the injected fluid will retain higher sub-cooling than would be observed in the test facility. To assess the impact of downcomer sub-cooling on the transient simulations, a sensitivity study was performed with NOTRUMP in which the intact DVI fluid streams (CMT and accumulator) were heated to [ ]^{a,c}. The results indicate the divergence observed between the test and NOTRUMP simulation was significantly reduced although not totally eliminated (Figure E-16 and Figure E-17). This indicates that core inlet sub-cooling, or lack thereof, is partly responsible for the divergence between the NOTRUMP simulation and the test response.

The core collapsed level (Figure E-18) plot is in reasonable agreement with the test up to approximately ADS-2 actuation. However, they diverge between [ ]^{a,b} and [ ]^{a,c} seconds due to lack of two-dimensional downcomer modeling and heating of DVI injection flow as discussed above. The core behavior between the test observed and NOTRUMP predictions re-converge at approximately [ ]^{a,c} seconds. In both cases, the core level initially decreases as inventory is lost from the system. The levels increase following accumulator injection. Once the accumulators empty, the levels continue to increase as a result of CMT-2 injection. Following CMT-2 empty, an injection gap period is encountered. During this period, the core collapsed level slowly decreases until IRWST-2 injection occurs. Since the NOTRUMP simulation predicts a slightly early IRWST-2 injection compared to the test results, it exhibits an earlier recovery than observed in the test. However, the level response is similar between the simulation and test data. A comparison of the core average void fraction is provided as Figure E-19. This figure shows lower predicted void fractions during the same divergence period as described above; however, once the conditions re-converge at near [ ]^{a,c} seconds, the NOTRUMP simulation and test data are in reasonable agreement for the remainder of the transient.

The collapsed upper plenum level (Figure E-20) indicates that both NOTRUMP and the test simulation have a significant amount of fluid in this region. The upper plenum collapsed level response in the NOTRUMP simulation indicates more sensitivity to the injection gap period than observed in the test (that is, NOTRUMP predicting a higher inventory loss compared to the test over the injection gap period). The upper plenum two-phase level (Figure E-21) follows the same trends as observed in the core and downcomer, that being that the trends are followed reasonably well until ADS-2 through about [ ]^{a,c} seconds. The two-phase level information indicates that both the test and NOTRUMP simulations behave similarly during the injection gap period with both the test and simulation indicating a decrease in mixture level until IRWST-2 injection commences. The NOTRUMP simulation and test data are considered to be in reasonable agreement.

Figure E-22 shows the collapsed liquid level in the downcomer for the test and the NOTRUMP simulation. Again there is reasonable agreement between the test and the simulation up to ADS-2 actuation. Following ADS-2 actuation, the test-observed and NOTRUMP-predicted behavior, while similar in trend, diverge. This is once again attributed to the lack of two-dimensional capability in the NOTRUMP [ ]^{a,c} downcomer (see Reference E1 for additional details) and downcomer sub-cooling as described previously. As such, the downcomer levels are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are considered to be reasonable beyond [ ]^{a,c} seconds.

These comparisons demonstrate that the highly ranked PIRT items related to the levels in the core, upper plenum, and downcomer are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are once again considered reasonable beyond [ ]^{a,c} seconds. The discrepancy period is not considered to be a serious deficiency as the vessel inventory at the critical time of intact IRWST injection is reasonably predicted by NOTRUMP and is consistent with past observations for the DVI line break (Reference E1).

Figure E-23 presents a comparison of the vessel mixture inventory between the test and NOTRUMP simulation. As can be seen, the NOTRUMP simulation generally underpredicts the test data with the exception of the period of divergence between ADS-2 and [ ]^{a,c} seconds. This indicates that during the time region of importance, (that is, post ADS-4 to IRWST injection) that the NOTRUMP code conservatively predicts the vessel conditions. The slightly early IRWST-2 injection, predicted by NOTRUMP, is clearly seen in this figure as the point at which the minimum inventory is predicted. This indicates that the NOTRUMP code is performing reasonably.

Figure E-24 shows the integrated mass flow through ADS stage-4 for the test and the NOTRUMP simulation. These curves show that the ADS stage-4 flow is slightly overpredicted by NOTRUMP after about 450 seconds. The flows match reasonably well as indicated by the parallel behavior of the integrated flow curves and the observed trends. This agreement in the slope of the curves demonstrates that the PIRT highly ranked items related to ADS stage-4 (critical flow, two-phase pressure drop, and valve loss coefficients) are predicted reasonably by NOTRUMP.

Figure E-25 shows the integrated mass flow out of the break for the test and the NOTRUMP simulation. For this simulation, the NOTRUMP model applied a discharge coefficient of [ ]^{a,b} to more accurately represent the results observed in the test. Differences in modeling of the break and break measurement system in the test and NOTRUMP simulations can also affect the results. This is described in more detail in the response to RAI.440.721(d) (Reference E1). Although the integrated break flow is slightly overpredicted by NOTRUMP, the general trends of the test break flow are similar to the prediction. This demonstrates that the PIRT highly ranked item of break critical flow can be predicted by NOTRUMP.

Figure E-26 and Figure E-27 show the total DVI line flow rates between the NOTRUMP simulation and the test for DVI line 1 and DVI line 2 respectively. The simulation data, provided for DVI line 1, represents the break flow from the DVI side piping of the DEDVI break. As can be seen, although the trends are predicted, the behavior of the ruptured DVI line (DVI-1) overpredicts the initial CMT draindown rate as described earlier. This causes an early prediction of ADS actuation for the NOTRUMP simulation compared to the test prediction. However, this is assessed to have a minimal impact on the results. As such, the results are considered reasonable. Figure E-27 presents the intact side DVI line flow

(DVI-2) for both the test and NOTRUMP simulation. The results indicate that the intact side DVI flow is predicted well by NOTRUMP. Since this path represents the makeup source, it represents an important characteristic that is well predicted by the NOTRUMP simulation.

Figure E-28 and Figure E-29 present the core inlet and core outlet temperatures between the test and NOTRUMP simulation respectively. The core inlet temperature is approximately the same as the simulation until approximately 150 seconds of the transient, while the outlet temperature is predicted well. After 300 seconds, the core inlet fluid temperature is overpredicted and is likely due to the removal of the PRHR model from the NOTRUMP simulation to conservatively account for the potential accumulation of non-condensable gases in the PRHR tubes, which cannot be directly modeled with NOTRUMP. As such, the NOTRUMP comparisons are considered reasonable.

E1.2 COMPARISON OF NOTRUMP SIMULATION TO TEST DATA FOR TEST DBA-03

Figure E-30 and Figure E-31 compare the pressure at the top of the pressurizer and downcomer regions for the test and the NOTRUMP simulation. The pressure decreases initially due to the blowdown through the break. The depressurization rate slows (and stops for NOTRUMP) when the primary system becomes saturated. Following actuation of ADS-1 at [ ]^{a,b} seconds in the test (84.4 seconds for NOTRUMP), the depressurization rate increases significantly. NOTRUMP predicts a higher pressure than observed in the test for most of the time. The trends observed in the pressurizer are also observed in the downcomer pressure response as well. As such, the agreement between the test data and the prediction is considered to be reasonable for the primary pressure response with the trends of the data being similar to that observed in the test.

Figure E-32 shows the collapsed liquid level in the pressurizer for the test and the NOTRUMP simulation. The break flow causes a rapid decrease in pressurizer level and empties the pressurizer at approximately [ ]^{a,b} seconds for the test and 70 seconds for the NOTRUMP simulation. The pressurizer level increases following ADS actuation for both the test and the simulation with NOTRUMP initially refilling faster than the test until ADS-2 actuation. The NOTRUMP simulation collapsed mixture level recovers to approximately the same level following ADS actuation. Following ADS-4 actuation, both the test and NOTRUMP simulations indicate a period of continued pressurizer refill until the ADS-4 flow paths become the dominant depressurization paths. The pressurizer collapsed level decreases in a similar manner following ADS-4 actuation for both the simulation and the test data. Therefore, the NOTRUMP results are considered to be in reasonable agreement with the test data.

Figure E-33 and Figure E-34 show the collapsed liquid levels in CMT-1 and CMT-2 for the test and the NOTRUMP simulation respectively. In the test, CMT-1, which is attached to the broken DVI line, begins draining out the break at [ ]^{a,b} seconds in the test compared to 20 seconds for the NOTRUMP simulation. This can also be seen in the CMT injection flow plots (Figure E-35 and Figure E-36). As such, the NOTRUMP simulation transitions from re-circulation to draindown mode earlier than observed in the test and, subsequently, predicts higher injection flows. The comparisons indicate that the NOTRUMP intact CMT drains earlier than observed in the test. This is due to both the earlier predicted emptying of the intact accumulator by the NOTRUMP simulation (Figure E-42 and Figure E-52) and the earlier IRWST injection observed in the test. The conclusions that can be reached are that the NOTRUMP results for the CMT behavior are considered to be in reasonable agreement with the test data.

The underprediction of the transition to CMT-1 draindown mode negligibly impacts the predicted ADS-1 actuation time compared to the test results and is considered reasonable.

Figure E-37 through Figure E-40 present the collapsed steam generator level comparisons between the test and NOTRUMP simulations. As can be seen, the NOTRUMP results are in good agreement with the test data.

Figure E-41 and Figure E-42 show the collapsed liquid levels in accumulator 1 and accumulator 2 for the test and the NOTRUMP simulation. As can be seen, the intact accumulator injection characteristics differ significantly between the NOTRUMP simulation and the test. When one reviews the injection characteristics compared to test DBA-02, the intact accumulator differs in an unexpected fashion. Since the differences between test DBA-02 and test DBA-03 are limited to the ADS-4 failure location, the changes expected, between test DBA-02 and DBA-03, should occur following ADS-4 actuation. However, as can be seen the transients diverge prior to this time. The comparison between the test and the NOTRUMP simulation is considered good for accumulator 1; however, the comparison for accumulator 2 is considered minimal. The minimal prediction is considered to have a negligible impact on the results as the composite effect of CMT and accumulator injection is reasonably/conservatively predicted by NOTRUMP. This is particularly evident in the time period prior to IRWST injection during which the NOTRUMP vessel mass is conservatively predicted relative to the test (Figure E-48).

The next series of plots relate to the collapsed and two-phase levels at different locations in the vessel. The trends of the simulation plots are in reasonable agreement with the test up to approximately ADS-2 actuation. Following ADS-2 actuation, the test and NOTRUMP simulations diverge as a result of the test-observed, two-dimensional downcomer behavior, which cannot be modeled with the NOTRUMP [ ]^{a,c} downcomer (see Reference E1 for additional details) and the downcomer sub-cooling as described in the previous discussion of test DBA-02.

The core collapsed level (Figure E-43) plot is in reasonable agreement with the test up to approximately ADS-2 actuation. However, they diverge between [ ]^{a,b} and [ ]^{a,c} seconds as discussed above. The core behavior between the test observed and NOTRUMP predictions re-converge at approximately [ ]^{a,c} seconds. In both cases, the core level initially decreases as inventory is lost from the system. The levels increase following accumulator injection. Once the accumulators empty, the levels continue to increase as a result of CMT-2 injection. For this case, the test indicates that continuous injection will occur while the NOTRUMP simulation indicates an injection gap period will occur. During this predicted injection gap, the NOTRUMP core mixture level decreases slightly until IRWST-2 injection occurs at which time a core level recovery occurs. A comparison of the core average void fraction is provided as Figure E-44. This figure shows lower predicted void fractions during the same divergence period as described above; however, once the conditions re-converge at near [ ]^{a,c} seconds, the NOTRUMP simulation and test data are in reasonable agreement for the remainder of the transient.

The collapsed upper plenum level (Figure E-45) indicates that both NOTRUMP and the test simulation have a significant amount of fluid in this region as was observed in test DBA-02. The upper plenum collapsed level response in the NOTRUMP simulation indicates the same behavior as observed in the test. The upper plenum two-phase level (Figure E-46) follows the same trends as observed in the core and downcomer, that being that the trends are followed reasonably well until ADS-2 through about [ ]^{a,c} seconds. The comparisons also indicate that that the NOTRUMP simulation does not recover the

two-phase level as quickly as a result of the predicted injection gap as compared to the test observed conditions. Following IRWST injection, the test indicates a more rapid increase in the two-phase mixture level as compared to the NOTRUMP simulation, which increases more slowly. As such, the two-phase mixture level is conservatively predicted by NOTRUMP following intact IRWST injection and is considered reasonable.

Figure E-47 shows the collapsed liquid level in the downcomer for the test and the NOTRUMP simulation. Again, there is reasonable agreement between the test and the simulation up to ADS-2 actuation. Following ADS-2 actuation, the test-observed and NOTRUMP-predicted behavior diverges. This is attributed to the lack of two-dimensional capability in the NOTRUMP [ ]^{a,c} downcomer (see Reference E1 for additional details) and downcomer sub-cooling as described previously. As such, the downcomer levels are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are considered to be reasonable beyond [ ]^{a,c} seconds.

These comparisons demonstrate that the highly ranked PIRT items related to the levels in the core, upper plenum, and downcomer are predicted reasonably well by NOTRUMP up to ADS-2 actuation and with the noted discrepancy between ADS-2 and [ ]^{a,c} seconds. The comparisons are once again considered reasonable beyond [ ]^{a,c} seconds.

Figure E-48 presents a comparison of the vessel mixture inventory between the test and NOTRUMP simulation. As can be seen, the NOTRUMP simulation generally underpredicts the test data with the exception of the period of divergence between ADS-2 and [ ]^{a,c} seconds. This indicates that during the time region of importance (that is, Post ADS-4 to IRWST injection), that the NOTRUMP code conservatively predicts the vessel conditions. The delay in the predicted IRWST-2 injection is clearly seen in this figure as the point at which the minimum inventory is predicted. This indicates that the NOTRUMP code is performing reasonably.

Figure E-49 shows the integrated mass flow through ADS stage-4 for the test and the NOTRUMP simulation. These curves show that the ADS stage-4 flow is slightly overpredicted by NOTRUMP. This comparison demonstrates that the PIRT highly ranked items related to ADS stage-4 (critical flow, two-phase pressure drop, and valve loss coefficients) are predicted reasonably by NOTRUMP.

Figure E-50 shows the integrated mass flow out of the break for the test and the NOTRUMP simulation. For this simulation, the NOTRUMP model applied a discharge coefficient of [ ]^{a,b} to more accurately represent the results observed in the test. Differences in modeling of the break and break measurement system in the test and NOTRUMP simulations can also affect the results. This is described in more detail in the response to RAI.440.721(d) (Reference E1). The test observed conditions indicate additional liquid discharge occurring at approximately [ ]^{a,c} seconds as a result of the higher observed downcomer mixture level compared to the NOTRUMP predicted results. In addition, since the test indicates earlier IRWST injection, compared to the NOTRUMP simulation, the break flows follow this trend as well. The general trends of the test break flow are similar to the NOTRUMP prediction. This demonstrates that the PIRT highly ranked item of break critical flow can be reasonably predicted by NOTRUMP.

Figure E-51 and Figure E-52 show the total DVI line flow rates between the NOTRUMP simulation and the test for DVI line 1 and DVI line 2, respectively. The simulation data, provided for DVI line 1,

represents the break flow from the DVI side piping of the DEDVI break. As can be seen, although the trends are predicted, the behavior of the ruptured DVI line (DVI-1) overpredicts the initial CMT draindown rate. This results in an early prediction of ADS actuation for the NOTRUMP simulation compared to the test prediction. However, this is assessed to have a minimal impact on the results. As such, the results are considered reasonable. Figure E-52 presents the intact side DVI line flow (DVI-2) for both the test and NOTRUMP simulation. The results indicate that the intact side DVI flow is predicted reasonably by NOTRUMP.

Figure E-53 and Figure E-54 present the core inlet and core outlet temperatures between the test and NOTRUMP simulation respectively. The core inlet temperature is approximately the same as the simulation until approximately 150 seconds of the transient, while the outlet temperature is predicted well. After 300 seconds, the core inlet fluid temperature is overpredicted and is likely due to the removal of the PRHR model from the NOTRUMP simulation to conservatively account for the potential accumulation of non-condensable gases in the PRHR tubes, which cannot be directly modeled with NOTRUMP. As such, the NOTRUMP comparisons are considered reasonable for the modeling capability available.

E1.3 NOTRUMP PREDICTED CORE VOID FRACTION DURING ACCUMULATOR INJECTION PERIOD

The purpose of this section is to demonstrate that the observed core void fraction misprediction during the accumulator injection period for the NOTRUMP simulations of the APEX-1000 DEDVI tests does not propagate to the IRWST injection phase of the transient.

To make this demonstration, NOTRUMP simulations were performed of the APEX-1000 DBA-02 test in which downcomer conditions were altered. In Section E1.1, a description of the NOTRUMP misprediction following ADS-2 actuation was provided. The divergence is attributed to two-dimensional downcomer behavior, which cannot be accurately modeled with the NOTRUMP [ ]^{a,c} downcomer. A review of the core inlet conditions (Figure E-28) indicates that NOTRUMP predicts subcooled inlet conditions; whereas, the test indicates saturated core entry conditions. This was attributed to several factors: the lack of multi-dimensional downcomer modeling capability, heating of the DVI injection flow as it impinges on the core barrel, and/or the underprediction of DVI injection condensation. The divergence occurs during the accumulator injection period. To assess the impact of this misprediction, sensitivity studies were performed.

A sensitivity study is provided in Section E1.1 in which the initial temperature of the intact DVI fluid streams (CMT and accumulator) were heated to [ ]^{a,c}. The results indicate a significant reduction in the observed divergence although it was not totally eliminated (see Figures E-16 and E-17). Subsequent efforts focused on forcing the NOTRUMP simulations to more closely match the observed test behavior during the accumulator injection period to determine if the misprediction affects the long-term behavior of the transient. This was accomplished by enhancing downcomer condensation along with the use of an alternate core drift flux model.

To best match the test simulation, the downcomer condensation was enhanced while the core drift flux models were altered from the Evaluation Model (EM) Yeh correlation to the EPRI drift flux model from transient initiation through most of the accumulator injection period. Figures E-55 through E-57 present

core average void fraction, intact DVI injection flow, and vessel mass comparison plots for APEX-1000 test DBA-02. As seen in Figure E-55, the core average void fraction behavior during the early phase of the transient is more closely simulated without significantly altering the intact DVI injection flow (Figure E-56). Finally, the vessel inventory comparison presented in Figure E-57 indicates the NOTRUMP simulation reasonably predicts the observed test behavior. Thus, the correction of the initial misprediction during the accumulator injection period does not significantly alter the simulation response after the accumulator injection period.

E2.0 OVERALL CONCLUSIONS

The following conclusions can be reached by reviewing the NOTRUMP predicted response compared to the test observed conditions:

- NOTRUMP predicts the effect of the ADS-4 single failure location as observed in the test.
- NOTRUMP conservatively predicts vessel inventory during the ADS-4 to IRWST injection period.
- NOTRUMP predicts the pressurizer mixture level performance reasonably well.
- NOTRUMP predicts IRWST injection flow reasonably well.
- The divergence of vessel inventory between ADS-2 actuation to approximately [ ]^{a,c} seconds is a multi-dimensional effect and sub-cooling effect, which cannot be properly modeled by NOTRUMP; however, the duration of this period is small and the ADS-4 to IRWST injection period is considered to be reasonable.
- The results indicate that the NOTRUMP code performs reasonably compared to tests designed specifically for comparisons to the AP1000 plant design. As such, it continues to be applicable for analyses of SBLOCA events for the AP1000 plant design.

E3.0 REFERENCES

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E2. OSU-APEX-03001, Revision 0, Scaling Assessment for the Design of the OSU APEX-1000 Test Facility, J. Reyes, et al., Oregon State University, Department of Nuclear Engineering, May 12, 2003.

E3. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 2, Section 8.0, August 1998.

E4. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, Appendix A, RAI-440.796F, Part a, August 1998.

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E6. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-339, August 1998.

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E8. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-721(f), August 1998.

a,c

Figure E-0 APEX NOTRUMP Original Noding Diagram

a,c

Figure E-1 APEX Pressurizer Model Modifications

a,c

[ ]

Figure E-2 APEX Core Makeup Tank Model Modifications

Table E1 DBA-02: Double-Ended Injection Line Break with Single Failure in ADS 4-2 Path Sequence of Events

Event	Test Data Time (seconds)[a,c]	NOTRUMP Time (seconds)
Break opens	[ ]	0.0
Reactor trip signal		0.0
Steam turbine stop valves close		0.0
CMT Isolation Valves Open		8.2
Main feed isolation valves begin to close		3.1
Reactor coolant pumps start to coast down		8.2
		8.2
		8.2
		8.2
ADS Stage 1		81.33
Intact accumulator injection starts		122
ADS Stage 2		128.33
ADS Stage 3		188.33
ADS Stage 4-1		246.33
ADS Stage 4-2		276.33
Intact accumulator empties		349.05
Intact loop core makeup tank empties		908
Intact loop IRWST injection starts*		1122
		1150*

Note:
*Continuous injection period

Table E2 DBA-03: Double-Ended Injection Line Break w/ Single Failure in ADS 4-1 Path Sequence of Events

Event	Test Data Time (seconds)[a,c]	NOTRUMP Time (seconds)
Break opens		0.0
Reactor trip signal		0.0
Steam turbine stop valves close		0.0
CMT Isolation Valves Open		8.2
Main feed isolation valves begin to close		3.1
Reactor coolant pumps start to coast down		8.2
		8.2
		8.2
		8.2
ADS Stage 1		84.22
Intact accumulator injection starts		123
ADS Stage 2		131.22
ADS Stage 3		191.22
ADS Stage 4-2		249.23
ADS Stage 4-1		279.23
Intact accumulator empties		346.44
Intact loop core makeup tank empties		922
Intact loop IRWST injection starts*		930
		975*

Note:
*Continuous injection period

a,b,c

Figure E-3 Test DBA-02, Pressurizer Pressure Comparison

a,b,c

Figure E-4 Test DBA-02, Downcomer Pressure Comparison

a,b,c

Figure E-5 Test DBA-02, Pressurizer Collapsed Level

a,b,c

Figure E-6 Test DBA-02, CMT-1 Collapsed Liquid Level

a,b,c

Figure E-7 Test DBA-02, CMT-2 Collapsed Liquid Level

a,b,c

Figure E-8 Test DBA-02, CMT-1 Injection Flow

a,b,c

Figure E-9 Test DBA-02, CMT-2 Injection Flow

a,b,c

Figure E-10 Test DBA-02, SG-2 Hot Side Collapsed Liquid Level

a,b,c

Figure E-11 Test DBA-02, SG-2 Cold Side Collapsed Liquid Level

a,b,c

Figure E-12 Test DBA-02, SG-1 Hot Side Collapsed Liquid Level

a,b,c

[ ]

Figure E-13 Test DBA-02, SG-1 Cold Side Collapsed Liquid Level

a,b,c

[ ]

Figure E-14 Test DBA-02, ACC-1 Collapsed Liquid Level

a,b,c

Figure E-15 Test DBA-02, ACC-2 Collapsed Liquid Level

a,b,c

Figure E-16 Test DBA-02, DC Sub-cooling Sensitivity, Core Collapsed Liquid Level

a,b,c

[ ]

Figure E-17 Test DBA-02, DC Sub-cooling Sensitivity Downcomer Collapsed Liquid Level

a,b,c

[ ]

Figure E-18 Test DBA-02, Core Collapsed Liquid Level

a,b,c

Figure E-19 Test DBA-02, Core Average Void Fraction

a,b,c

Figure E-20 Test DBA-02, Upper Plenum Collapsed Liquid Level

a,b,c

Figure E-21 Test DBA-02, Upper Plenum Two-Phase Mixture Level

a,b,c

Figure E-22 Test DBA-02, Downcomer Collapsed Liquid Level

a,b,c

Figure E-23 Test DBA-02, RPV Mixture Mass

a,b,c

Figure E-24 Test DBA-02, Integrated ADS-4 Discharge

a,b,c

[ ]

Figure E-25 Test DBA-02, Integrated Vessel Side Break Flow

a,b,c

[ ]

Figure E-26 Test DBA-02, DVI-1 Injection Flow (Loop Side Break)

a,b,c

Figure E-27 Test DBA-02, DVI-2 Injection Flow

a,b,c

Figure E-28 Test DBA-02, Core Inlet Temperature

a,b,c

Figure E-29 Test DBA-02, Core Outlet Temperature

a,b,c

Figure E-30 Test DBA-03, Pressurizer Pressure

a,b,c

Figure E-31 Test DBA-03, Downcomer Pressure

a,b,c

Figure E-32 Test DBA-03, Pressurizer Collapsed Liquid Level

a,b,c

[ ]

Figure E-33 Test DBA-03, CMT-1 Collapsed Liquid Level

a,b,c

[ ]

Figure E-34 Test DBA-03, CMT-2 Collapsed Liquid Level

a,b,c

Figure E-35 Test DBA-03, CMT-1 Injection Flow

a,b,c

Figure E-36 Test DBA-03, CMT-2 Injection Flow

a,b,c

[ ]

Figure E-37 Test DBA-03, SG-2 Hot Side Collapsed Liquid Level

a,b,c

[ ]

Figure E-38 Test DBA-03, SG-2 Cold Side Collapsed Liquid Level

a,b,c

Figure E-39 Test DBA-03, SG-1 Hot Side Collapsed Liquid Level

a,b,c

Figure E-40 Test DBA-03, SG-1 Cold Side Collapsed Liquid Level

a,b,c

Figure E-41 Test DBA-03, ACC-1 Collapsed Liquid Level

a,b,c

Figure E-42 Test DBA-03, ACC-2 Collapsed Liquid Level

a,b,c

Figure E-43 Test DBA-03, Core Collapsed Liquid Level

a,b,c

Figure E-44 Test DBA-03, Core Average Void Fraction

a,b,c

[ ]

Figure E-45 Test DBA-03, Upper Plenum Collapsed Liquid Level

a,b,c

[ ]

Figure E-46 Test DBA-03, Upper Plenum Two-Phase Mixture Level

a,b,c

Figure E-47 Test DBA-03, Downcomer Collapsed Liquid Level

a,b,c

Figure E-48 Test DBA-03, RPV Mixture Mass

a,b,c

Figure E-49 Test DBA-03, Integrated ADS 4 Discharge

a,b,c

Figure E-50 Test DBA-03, Integrated Break Discharge

a,b,c

Figure E-51 Test DBA-03, DVI-1 Injection Flow (Loop Side Break)

a,b,c

Figure E-52 Test DBA-03, DVI-2 Injection Flow

a,b,c

Figure E-53 Test DBA-03, Core Inlet Temperature

a,b,c

Figure E-54 Test DBA-03, Core Outlet Temperature

a,b,c

Figure E-55 Test DBA-02, Core Average Void Fraction

a,b,c

Figure E-56 Test DBA-02, Intact DVI Line Flow

a,b,c

[ ]

Figure E-57 Test DBA-02, Vessel Inventory

APPENDIX F

HOT LEG/UPPER PLENUM ENTRAINMENT SENSITIVITY STUDIES

F1.0 IDENTIFICATION OF LIMITING TRANSIENT

F1.1 IDENTIFICATION OF LIMITING SBLOCA VIA IMPORTANT ENTRAINMENT PARAMETER AT CORE EXIT

To determine the limiting transient in terms of upper plenum entrainment, non-dimensional gas velocities between several NOTRUMP simulations performed in the Design Control Document (DCD) were reviewed and compared. The non-dimensional gas velocity is an important parameter found in many entrainment correlations. Figures F-1 through F-4 provide non-dimensional gas velocity (j_g*) at the core exit as predicted by NOTRUMP for the AP1000 DCD cases. The comparisons show that for a given backpressure, the double-ended direct vessel injection (DEDVI) case is somewhat more limiting with respect to entrainment-related phenomena at the core exit than other DCD cases.

F1.2 IDENTIFICATION OF LIMITING SBLOCA VIA DOWNCOMER/REACTOR VESSEL INVENTORY TEST DATA

Given adequate core cooling, the next best indicator for identifying limiting small-break loss-of-coolant accident (SBLOCA) transients is downcomer/reactor vessel level or inventory. Figures F-5 through F-9 show downcomer/reactor vessel level data for the APEX-600 DEDVI (SB11 and SB12), 2-inch Cold Leg Break (SB01 and SB18), and Inadvertent Automatic Depressurization System (ADS) (SB14) tests. From these figures, it can be seen that the minimum reactor vessel liquid level (approximately [ ]^{a,c} inches) occurs in the DEDVI tests; minimum reactor vessel liquid levels for the other tests occur at levels approximately [ ]^{a,c} inches. The DEDVI event represents the most challenging DBA SBLOCA because it is a significant size break located low in the system, coupled with the most significant loss of safety injection capability due to a single failure. One-half of all safety injection capability is lost in the DEDVI event, and this is evidenced by the coincident low downcomer level (approximately [ ]^{a,c} inches). In the other accidents, the downcomer level is [ ]^{a,c} inches above the reactor vessel liquid level at the minimum condition. Based upon APEX-600 test experience and results obtained from APEX-1000 test/analysis, the DEDVI event represents the limiting downcomer/reactor vessel inventory for AP1000 SBLOCA design basis accident (DBA) events.

F1.3 CONCLUSIONS

Therefore, based upon downcomer/reactor vessel liquid inventory test data/analysis and evaluation of an important entrainment parameter, j_g*, at the core exit, the DEDVI event is the most limiting SBLOCA DBA event for the AP1000 for a given backpressure.

F2.0 SENSITIVITY MODELING

In order to assess the potential impact of upper plenum and hot leg entrainment on the AP1000 plant design, a sensitivity study was performed with the Advanced Plant version of the NOTRUMP computer code. The AP1000 plant model, as defined for the DCD analysis effort, was modified as follows to perform this study:

- Insertion of fluid node [ ]^{a,c} in the core fluid-node stack that represents the [ ]^{a,c} and the [ ]^{a,c}. This node is inserted between

fluid nodes [ ]^{a,c} and [ ]^{a,c} from the base DCD model. This results in the insertion of two additional flow links ([ ]^{a,c} and [ ]^{a,c}) for the added fluid node for the interior flow and reflux flow link types. In the base DCD model, this [ ]^{a,c} region is lumped into the [ ]^{a,c} stack. This modification was performed to improve the delineation between the [ ]^{a,c} regions. The base NOTRUMP noding diagram can be found in Figure F-10 with the noding modifications performed shown in Figure F-11.

- To account for a potential non-conservative pressure drop when modeling the fluid node/flow paths as homogenous, an additional pressure drop penalty, [ ]^{a,c}, is added to the ADS-4 discharge paths at the time when they become [ ]^{a,c}. This penalty corresponds to [ ]^{a,c}. This penalty is achieved via an [ ]^{a,c} as determined by the detailed momentum flux model (FLOAD4, Reference F1).

To confirm that the re-nodalization that was performed did not significantly impact the transient results, a comparison of the base DCD and the revised nodalization results will be presented first.

Figure F-12 presents the core/upper plenum region two-phase mixture level comparison between the base DCD model and the revised noding model. As can be seen, the changes between the two cases are negligible. Figure F-13 presents the two-phase downcomer mixture level comparison for the downcomer region between these same two cases. Again, the differences between the two cases are negligible.

Figure F-14 presents a comparison of the upper downcomer pressure between the two cases. As can be seen, the pressure responses are nearly identical. As a result, the predicted intact DVI line IRWST injection flow (Figure F-15) is also unchanged.

As can be observed from Figure F-16, the vessel mixture masses are comparable between the two cases with the revised noding case having a slightly lower overall vessel inventory. This is due primarily to the improved resolution of the vessel mass in the [ ]^{a,c}. The revised noding reflects a more accurate depiction of the geometry and subsequently the void profile in this region. However, this does not impact the active fuel region of the vessel as can be seen in Figure F-17 and Figure F-18 respectively. Both the active region mass and void fraction profiles are approximately the same. As a result, the active fuel region collapsed mixture level (Figure F-19) is also approximately the same.

Figure F-20 presents the Pressurizer level response for the two cases. As can be seen, the responses are nearly the same for both cases.

Figure F-21 and Figure F-22 present the ADS-4 Integrated liquid and vapor discharges for the two cases respectively. Again, the differences between these two cases are considered to be negligible.

Now that the baseline case has been established (i.e., re-noded core/upper plenum region), the sensitivity case can be described. The sensitivity case is performed to assess the effect of higher than expected entrainment in the upper plenum and hot legs on the overall system response and core cooling. The higher than expected entrainment is included in the analysis by assuming homogenous conditions in the upper plenum, hot legs and ADS-4 piping. The sensitivity involves the conversion of the fluid nodes

representing the Upper Plenum [ ]^{a,c}, Hot Legs [ ]^{a,c}, the PRHR Inlet [ ]^{a,c} to homogenous following ADS-4 actuation (~500 seconds). This also involves making the hot leg inlet flow paths [ ]^{a,c}, PRHR Inlet [ ]^{a,c}, and ADS-4 Inlet [ ]^{a,c} homogenous as well. As a result of the noding modifications, the Upper Plenum fluid node must be removed from the core fluid-node stack. This will allow for the formation of a distinct two-phase mixture level below the core plate should the conditions support it's generation. If this modification is not performed, the NOTRUMP mixture level tracking model would not allow the region below the core plate to form a distinct mixture level and subsequently, core uncovery would not occur unless the fluid node void fractions in the region became 1.0 (i.e., all vapor). In addition, since the homogenous treatment of this region will eliminate the pressure drop effect out of the fluid stored in the upper plenum, the NOTRUMP model was conservatively adjusted [ ]^{a,c}. This was accomplished by applying an additional [ ]^{a,c} flow paths.

Figure F-23 presents a comparison of the upper downcomer pressure between the base and sensitivity cases. As can be seen, the sensitivity case results in higher upper downcomer pressure and subsequently results in delayed IRWST injection (Figure F-24). This can also be observed in the intact DVI line flow, which comprises all intact injection flow components (i.e., Accumulator, CMT and IRWST) per Figure F-25. As expected, the initial ADS-4 liquid discharge is much higher (Figure F-26) until the inventory which resided in the upper plenum and hot leg regions was depleted (Figure F-27). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure F-28) and subsequently higher RCS pressures.

Due to the elimination of the inventory stored in the upper plenum, the downcomer mass is also reduced (Figure F-29) and is caused by the displacement of the upper plenum mixture. Since the static head that existed in the upper plenum is eliminated when the model is made homogenous, the downcomer mixture is subsequently driven into the core as the static heads equilibrate. This results in the core region mass increasing initially due to the introduction of cold downcomer fluid to the core region (Figure F-30). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure F-31); however, this inventory is sufficient for adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure F-26). Even though there is no liquid storage in the upper plenum for the homogenous case (Figure F-32), the core collapsed level (Figure F-33) is not impacted significantly.

The pressurizer mixture level response (Figure F-34) reflects the change in pressure response (Figure F-23) observed in the model as a result of the sensitivity study.

This sensitivity demonstrates that the AP1000 plant response is relatively insensitive to upper plenum and hot leg entrainment. Even with the assumption of homogenous fluid nodes above the core, adequate core cooling is demonstrated. Since no sustained core uncovery was predicted to occur, no rod heatup calculations were performed. In the event that sustained core uncovery would be predicted to occur, rod heatup calculations using appropriate methods (either adiabatic heatup calculations, or performance of specific rod heatup calculations with a pertinent calculational tool) will be performed to demonstrate the available margin to the 10 CFR 50.46 limits of peak cladding temperature less than 2200°F and cladding oxidation less than 17 percent.

F3.0 REFERENCES

F1. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, Appendix A, RAI-440.796F, Part a, August 1998.

F2. Response to DSER Open Item 25.1-3.

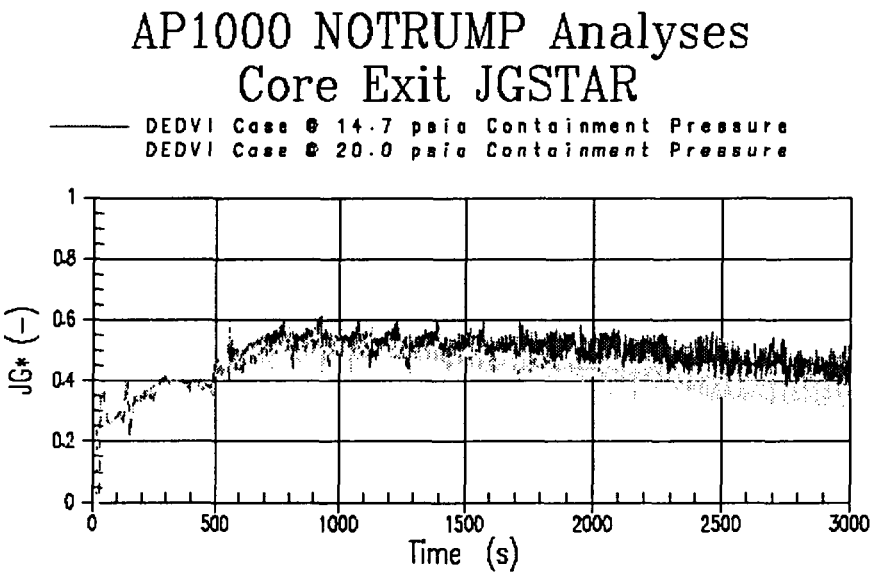


Figure F-1 AP1000 Core Exit j_g*, DEDVI 14.7 Versus 20.0 psi

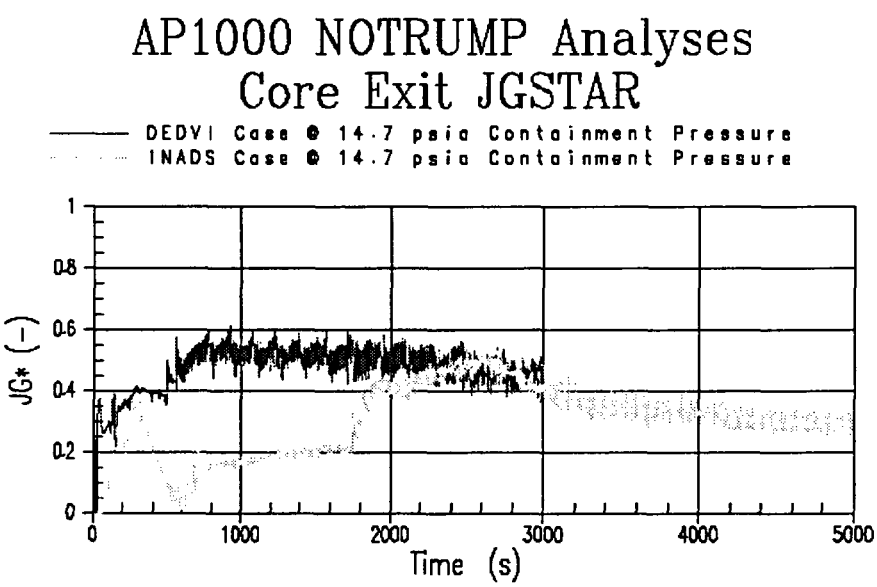


Figure F-2 AP1000 Core Exit j_g, DEDVI 14.7 Versus Inadvertent ADS 14.7**

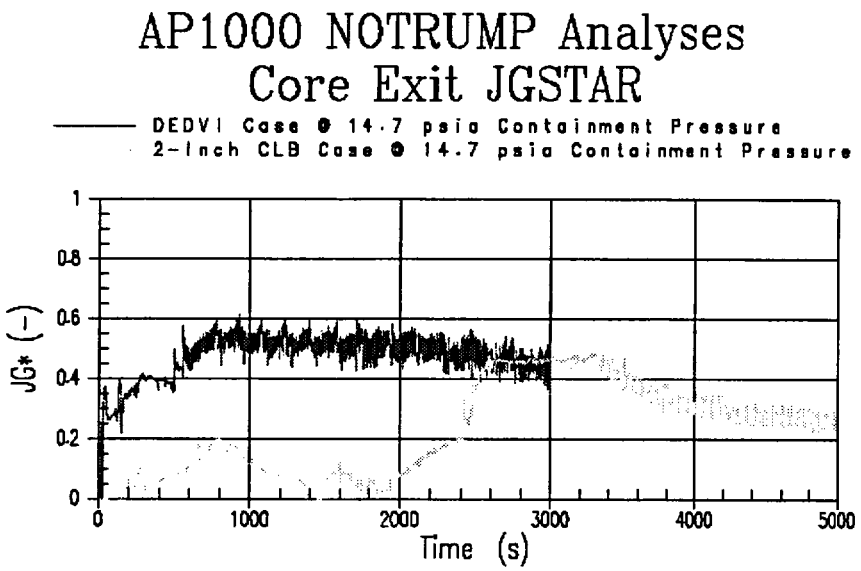


Figure F-3 AP1000 Core Exit j_g*, DEDVI 14.7 Versus 2-Inch Cold Leg Break 14.7**

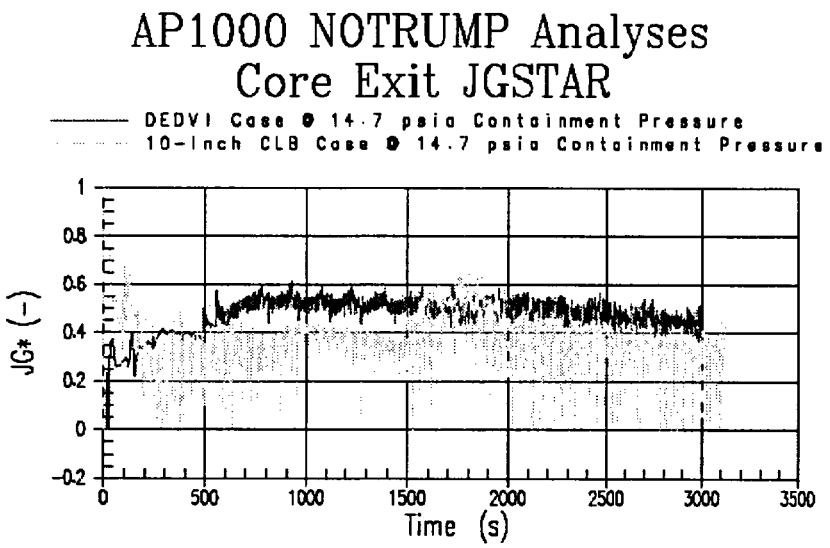


Figure F-4 AP1000 Core Exit j_g*, DEDVI 14.7 Versus 10-Inch Cold Leg Break 14.7**

a,c

Figure F-5 APEX-600 DEDVI Test SB11

a,c

Figure F-6 APEX-600 DEDVI Test SB12

a,c

Figure F-7 APEX-600 2-Inch Cold Leg Test SB01

a,c

Figure F-8 APEX-600 2-Inch Cold Leg Test SB18

a,c

Figure F-9 APEX-600 Inadvertent ADS Test SB14

a,c

Figure F-10 AP1000 NOTRUMP Noding Diagram

a,c

[ ]

Figure F-11 Vessel Noding Modifications

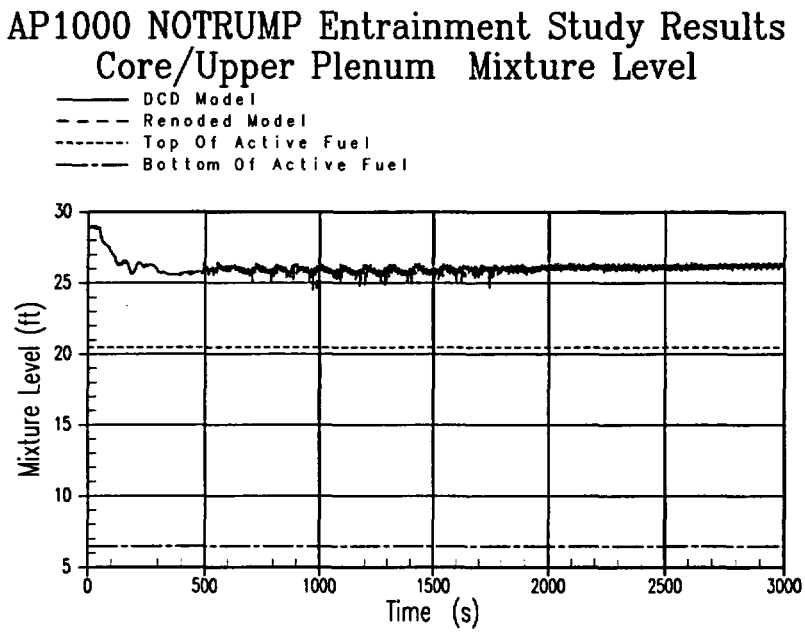


Figure F-12 Core/Upper Plenum Mixture Level Comparison

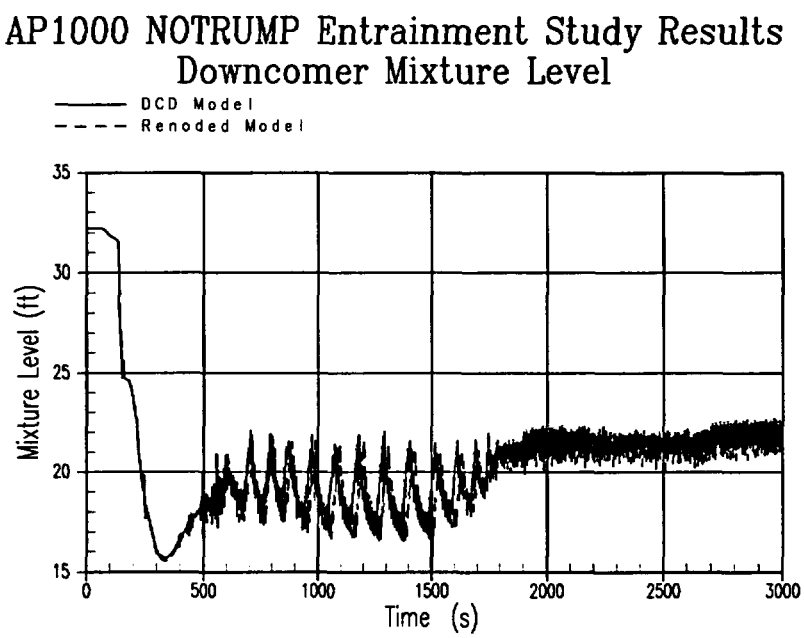


Figure F-13 Downcomer Level Comparison

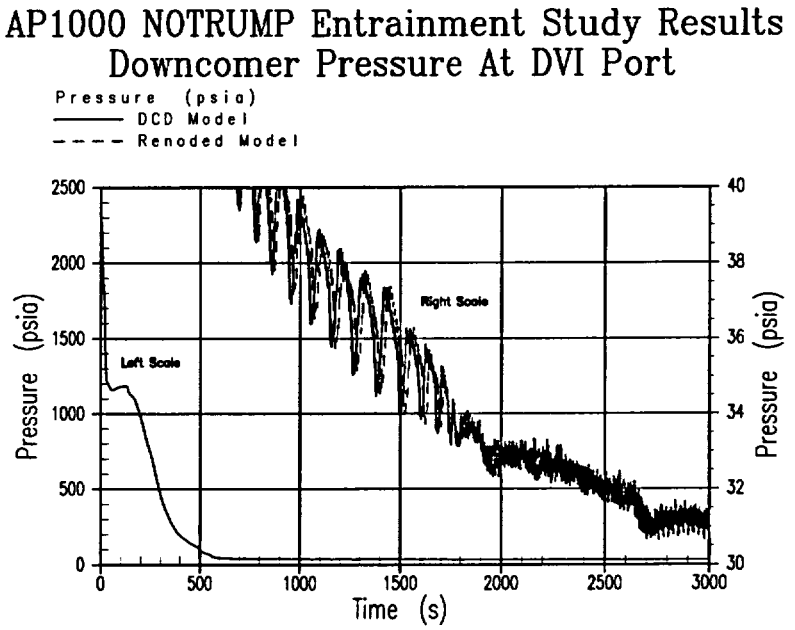


Figure F-14 Downcomer Pressure Comparison

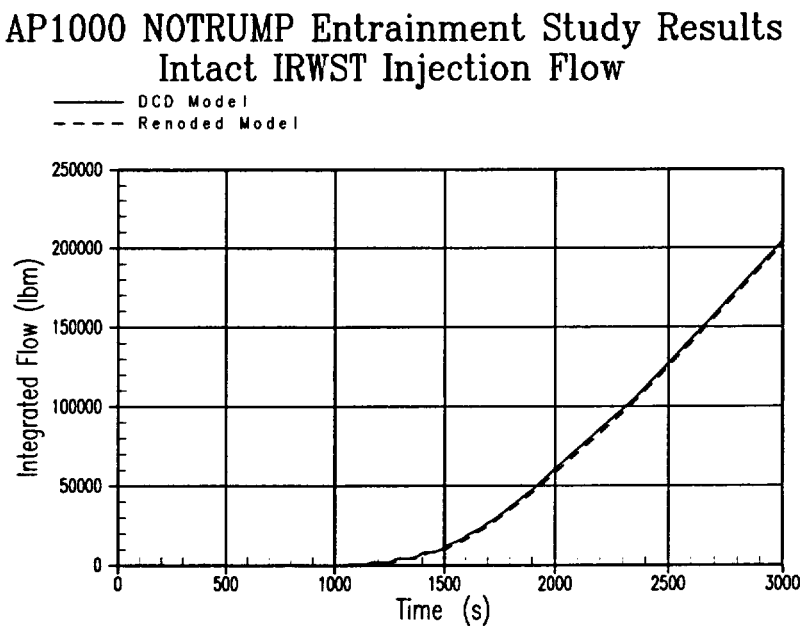


Figure F-15 Intact IRWST Injection Comparison

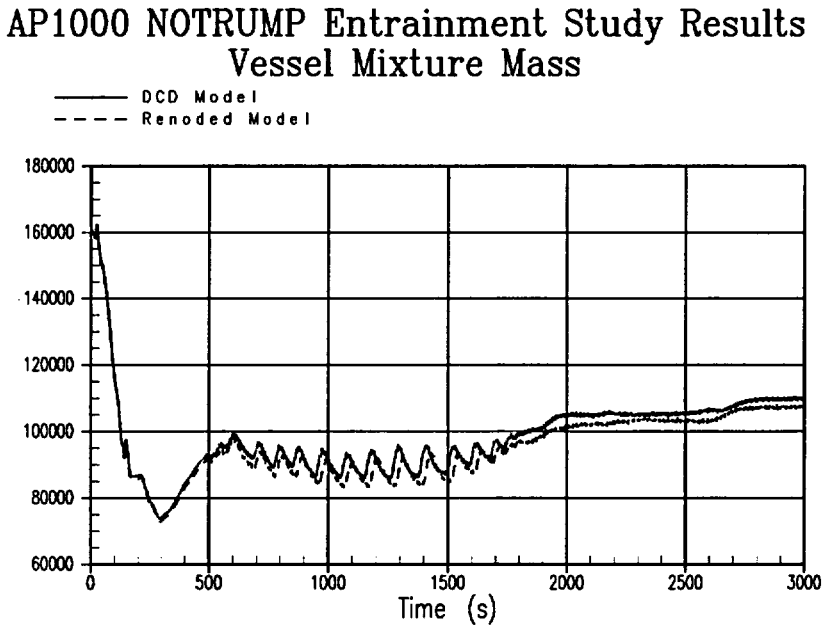


Figure F-16 Vessel Mixture Mass Comparison

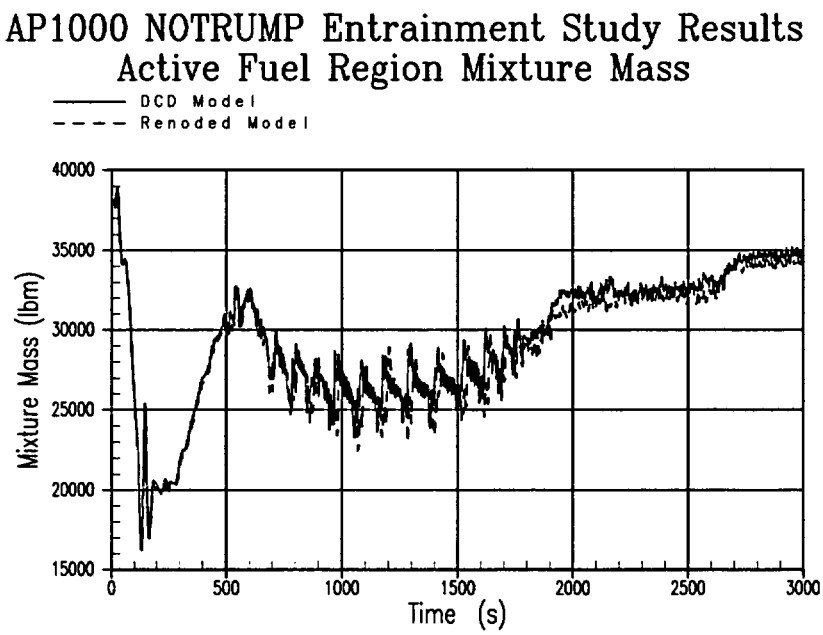


Figure F-17 Active Fuel Region Mixture Mass Comparison

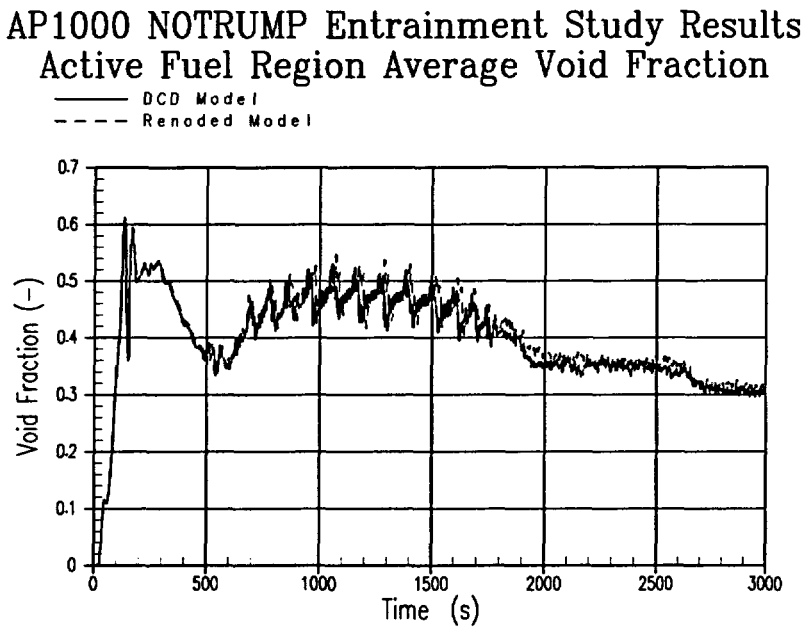


Figure F-18 Active Fuel Region Core Average Void Fraction

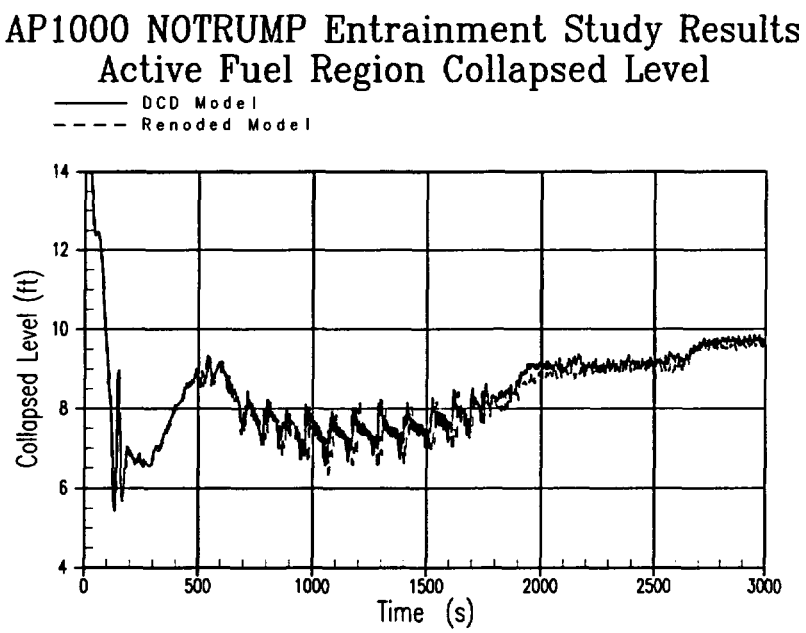


Figure F-19 Active Fuel Region Core Collapsed Level

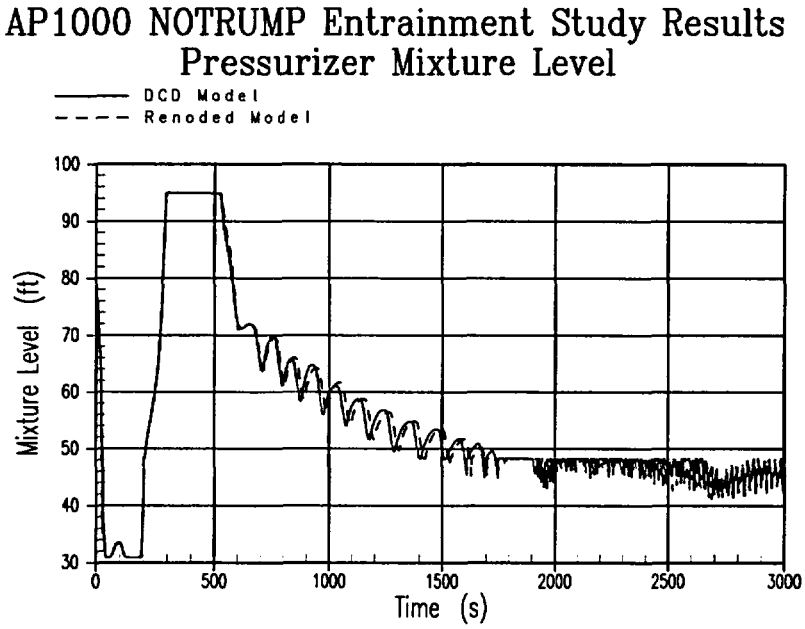


Figure F-20 Pressurizer Level Comparison

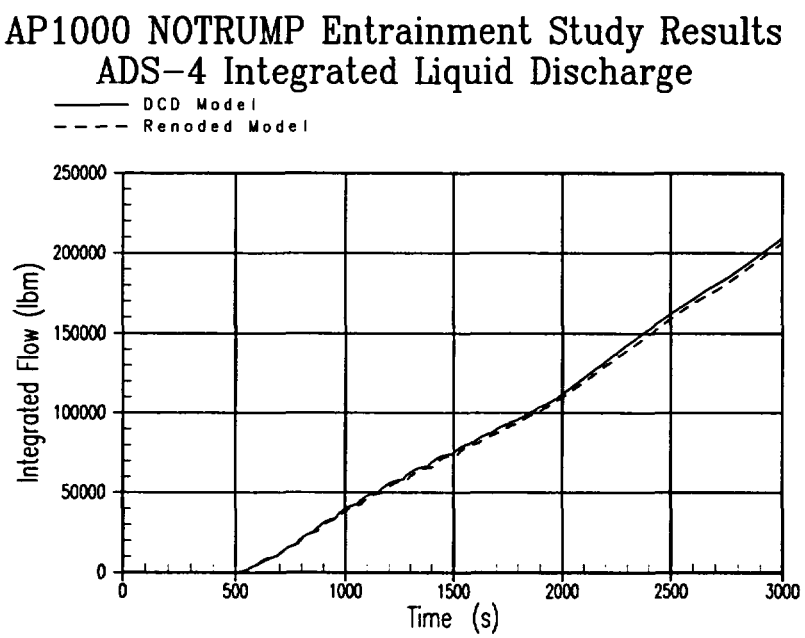


Figure F-21 ADS-4 Liquid Discharge Comparison

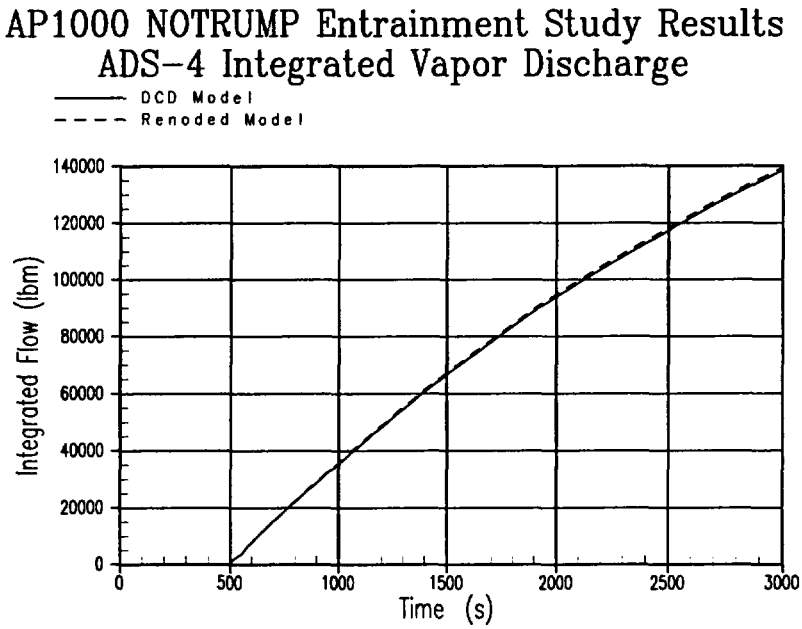


Figure F-22 ADS-4 Vapor Discharge Comparison

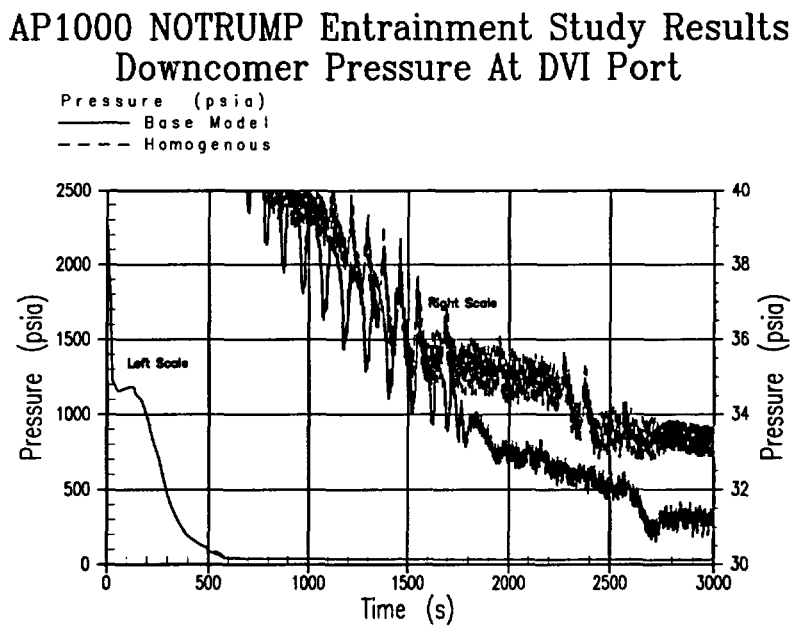


Figure F-23 Downcomer Pressure Comparison

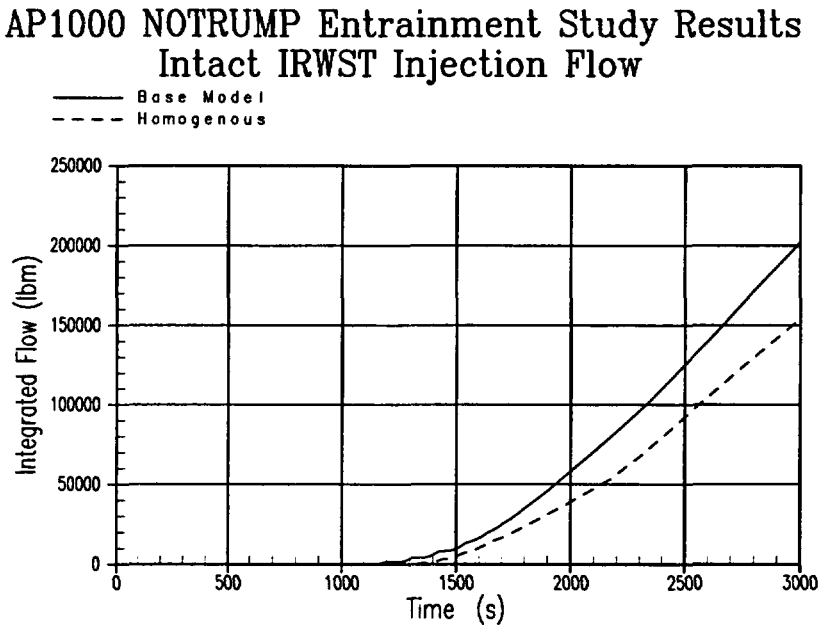


Figure F-24 Intact IRWST Injection Flow

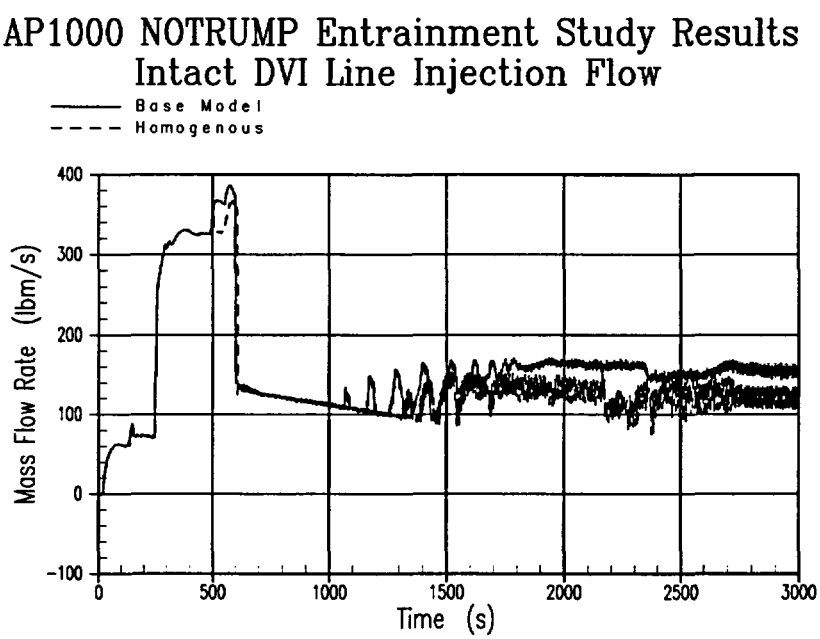


Figure F-25 Intact DVI Line Injection Flow

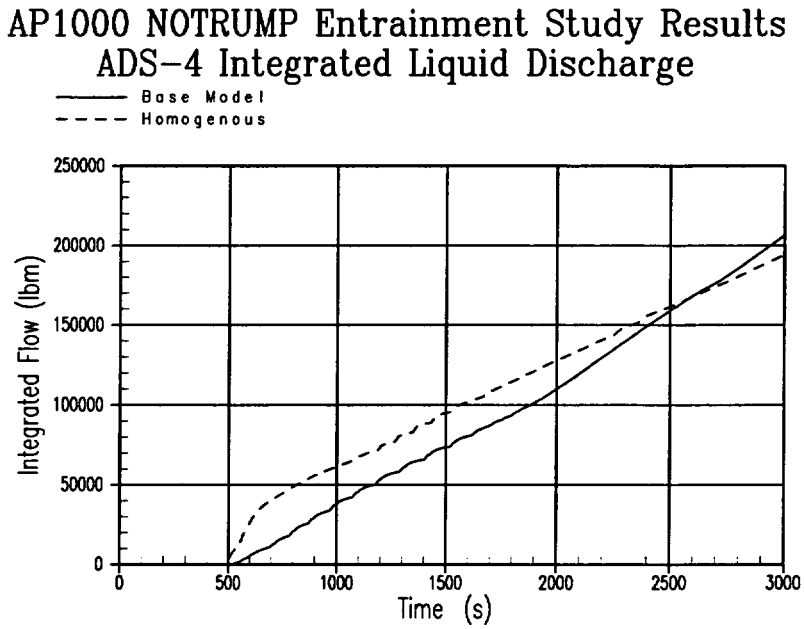


Figure F-26 ADS-4 Integrated Liquid Discharge Comparison

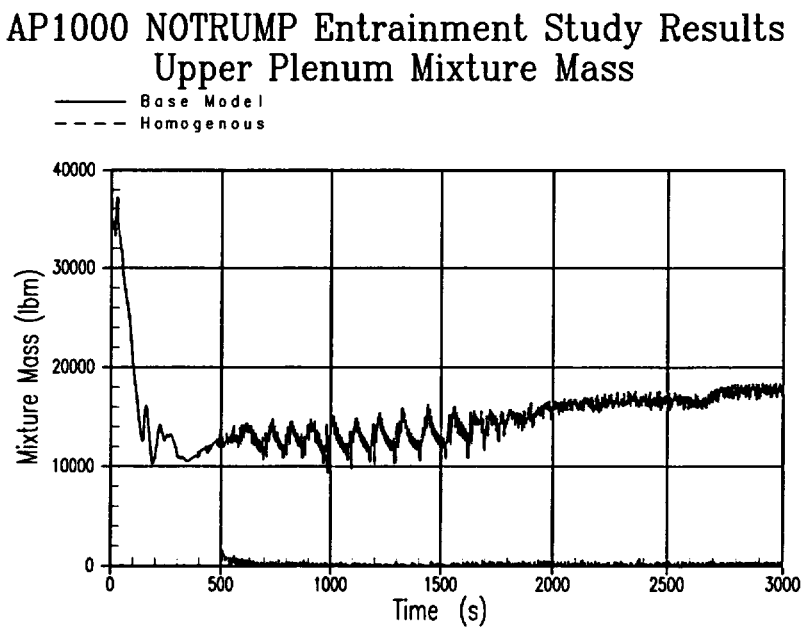


Figure F-27 Upper Plenum Mixture Mass Comparison

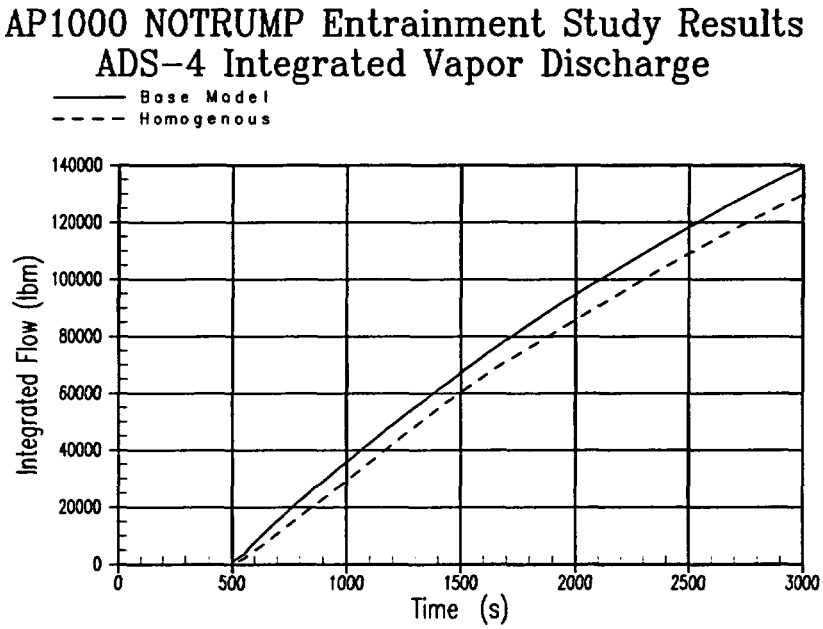


Figure F-28 ADS-4 Integrated Vapor Discharge Comparison

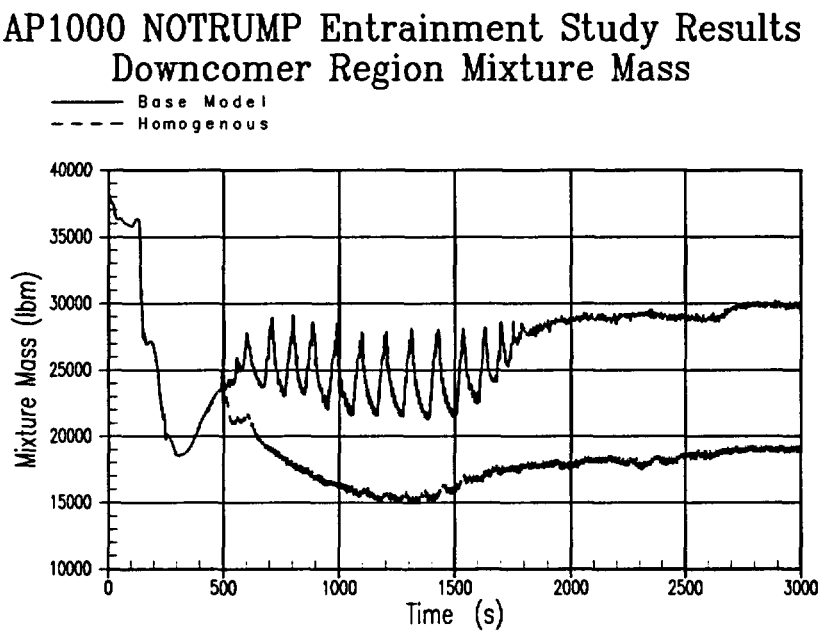


Figure F-29 Downcomer Region Mass Comparison

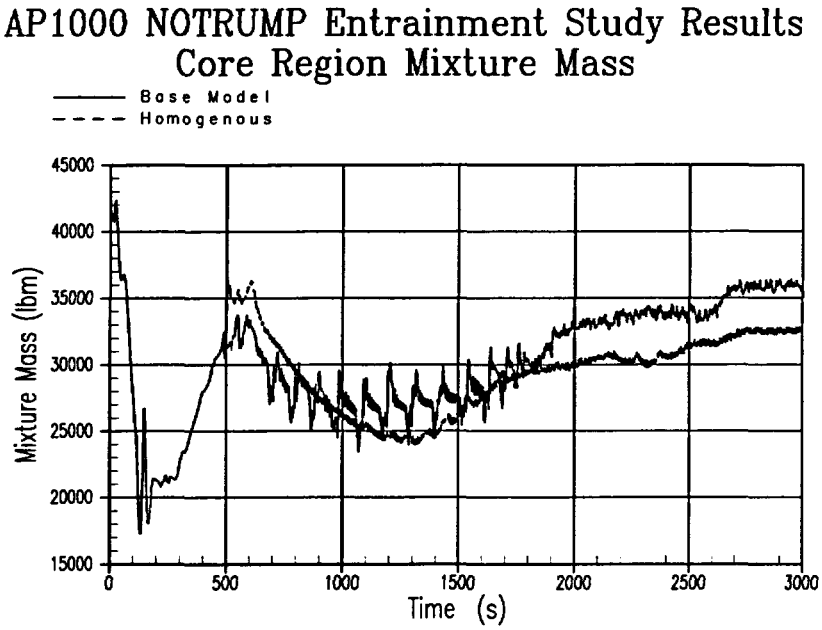


Figure F-30 Core Region Mass Comparison

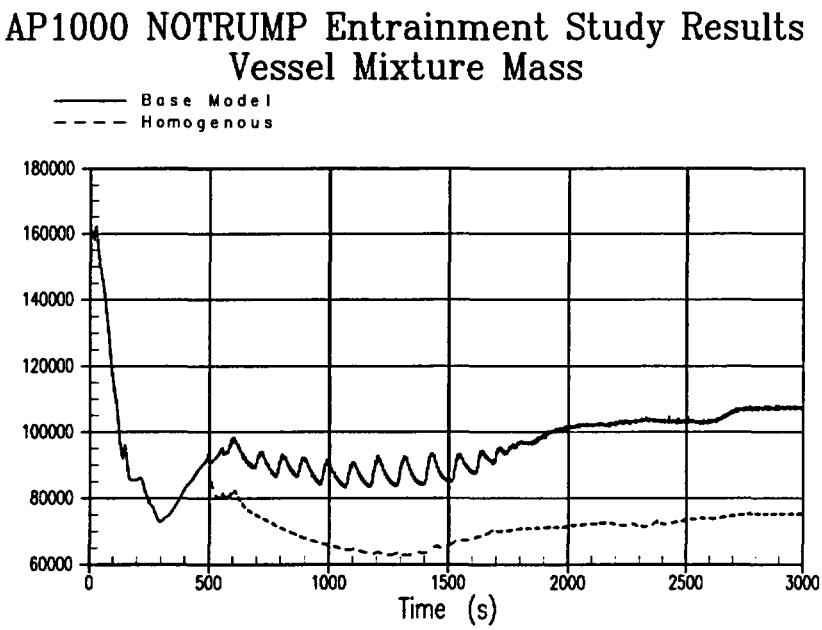


Figure F-31 Vessel Mixture Mass Comparison

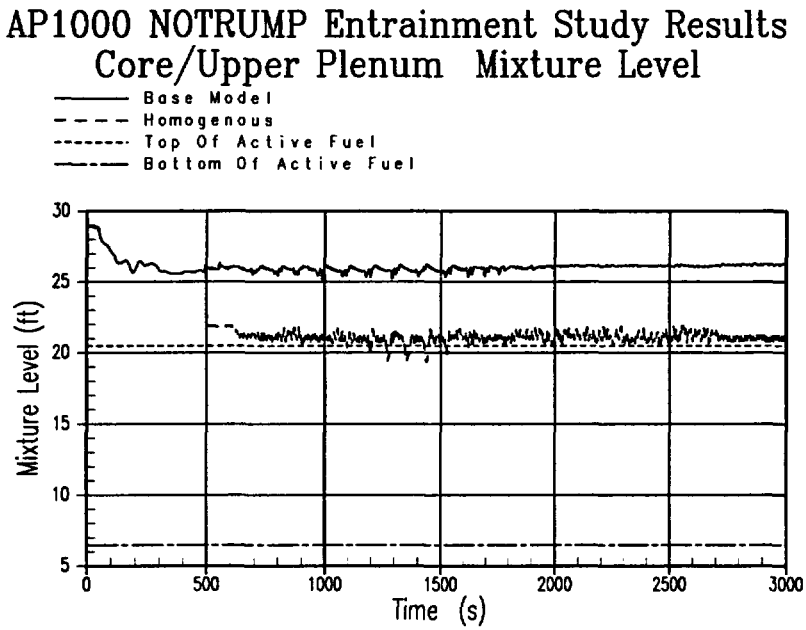


Figure F-32 Core/Upper Plenum Mixture Level Comparison

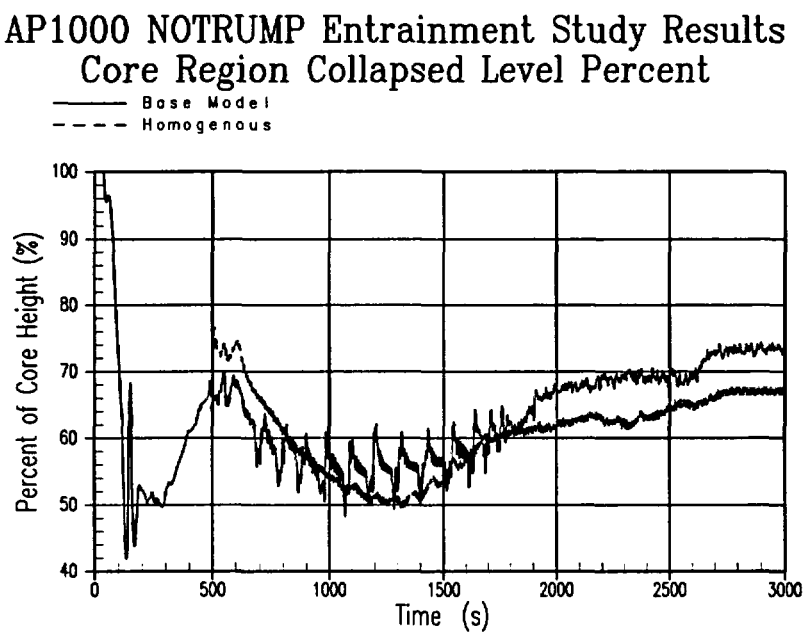


Figure F-33 Core Collapsed Level Comparison

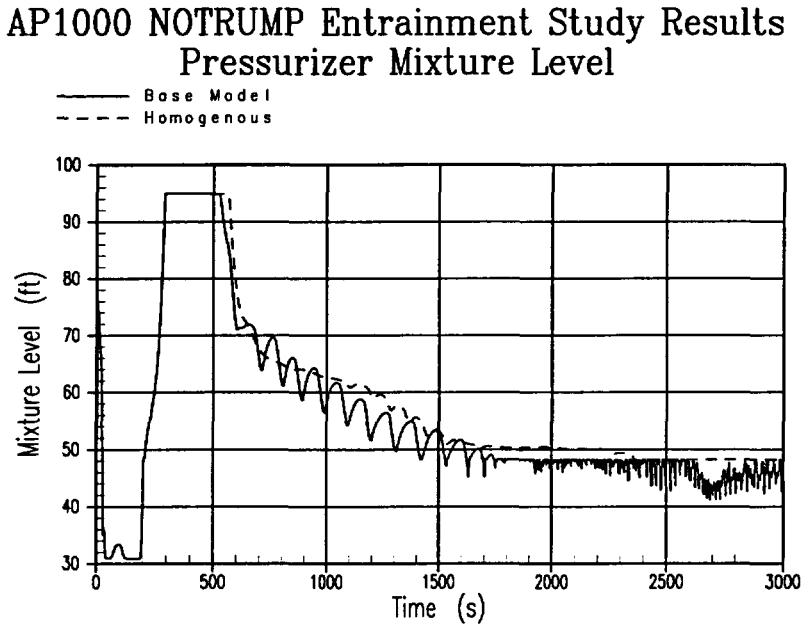


Figure F-34 Pressurizer Mixture Level Comparison

APPENDIX G

VALIDATION OF CORE VOID FRACTION MODEL USED IN NOTRUMP AGAINST FULL-SCALE DATA

The NOTRUMP core level swell model is based on the use of the Cunningham-Yeh void fraction correlation (Ref. G1) implemented as a drift flux model. The Cunningham-Yeh correlation was validated beyond its original data base by comparing its predictions with results from full-scale bundle experiments at conditions that are prototypical of the ADS-4/IRWST transition phase of the AP1000.

In particular, the following tests were considered:

- **FLECHT-SEASET**: Runs 35114, 31504, 31805, 31203, 34006
- **FLECHT-Skewed**: Runs 13404, 15606, 13609, 15713, 16022
- **G1**: Runs 28, 35, 38, 42, 43, 58, 59, 61
- **G2**: Runs 728, 729, 730, 732, 733
- **ACHILLES**: Runs AIL066, AIL069
- **THETIS**: Runs T2L101, T2L103, T2L098

Note that FLECHT-SEASET and FLECHT-Skewed are reflood tests. However, data was considered soon after the bundle is quenched when the power level, pressure, and bundle flow are more similar to the conditions expected in the AP1000 during the considered portion of the SBLOCA portion. All other tests are boil-off tests, which also have pressure and power conditions similar to the AP1000. On the other hand, in the boil-off tests, the liquid supply is insufficient to remove the power generated in the bundle. During the boil-off tests, the mixture level drops below the top of the heated section. Once the heated rods are exposed to the steam, an almost adiabatic heat-up occurs because of the degraded heat transfer in the region above the mixture level.

For the boil-off tests, data was extracted at different times when the mixture level was located in the upper portion of the bundle (8 to 12 feet from the bottom of the heated length).

Table G-1 shows the expected range of condition in the AP1000 and conditions for the tests that were selected for the additional validation of the Cunningham-Yeh model.

Table G-1 AP1000 and Full-Scale Tests Range of Conditions

Test	Pressure (psia)		Power (kW/ft)		Power Shape	Core/Assembly Flow (in/sec)		Inlet Subcooling (F)	
AP1000	20	45	0.02	0.18	Top Skewed	0.4	0.8	14	80
FLECHT-SEASET	39	40	0.13	0.24	Approx. Cos.	0.6	1.5	14	144
FLECHT-Skewed	21	41	0.25	0.42	Top Skewed	0.7	1.5	5	142
G1	15	15	0.09	0.26	Approx. Cos.	< 0.1	1.5	110	
G2	15	50	0.05	0.19	Approx. Cos.	0.1	0.6	0	10
								Subc. Length (ft)	
ACHILLES	17	29	0.10	0.10	Approx. Cos.	< 0.1		0.2	0.5
THETIS	32	32	0.02	0.17	Cosine	< 0.1		0.5	2.9

Note that for the THETIS and ACHILLES series, the effect of sub-cooling was directly reported in terms of sub-cooled length (Zsub) from the bottom of the heated length.

At a given time, for each test the vapor velocity was obtained as follow:

$$j_g(z) = \frac{P_{avg} N_{rod} \int_{z_{sub}}^{z} F(z)dz}{h_{fg} \rho_g A_b}$$

where:

P_{avg} = Average Rod Power kW/ft

N_{rod} = Number of Rods in Core/Assembly

$F(z)$ = Axial Power Shape

A_b = Flow Area Core/Assembly

Similarly, the liquid superficial velocity was calculated from a quasi-steady state mass balance by knowing the inlet flow at the given time. Knowing phasic superficial velocities, the void fraction axial distribution was obtained from the Cunningham-Yeh model:

$$\alpha(z) = 0.925 \left(\frac{\rho_g}{\rho_l}\right)^{0.239} \left(\frac{j_g(z)}{V_{bcr}}\right)^{b} \left(\frac{j_g(z)}{j_g(z) + j_l(z)}\right)^{0.6}$$

where:

$$V_{bcr} = \frac{2}{3}(gR_{bcr})^{0.5}$$

$$R_{bcr} = \left(\frac{1.53}{2/3}\right)^2 \left(\frac{\sigma}{g\rho_1}\right)^{0.5}$$

and

$$b = 0.67, \quad \text{if } \frac{j_g}{V_{bcr}} \le 1.0$$

$$b = 0.47, \quad \text{if } \frac{j_g}{V_{bcr}} > 1.0$$

The collapsed liquid level Z_{CLL} in the bundle was then calculated from:

$$Z_{CLL} = Z_{sub} + \int_{Z_{sub}}^{Z_{mix}} (1 - \alpha(z))\, dz$$

Where Z_{sub} and Z_{mix} were estimated from the test.

Finally the swell S was defined as follows:

$$S = \frac{Z_{mix} - Z_{sub}}{Z_{CLL} - Z_{sub}} = \frac{1}{1 - \overline{\alpha}}$$

The predicted swell S_c was then compared to the observed value S_m in Figure G-1.

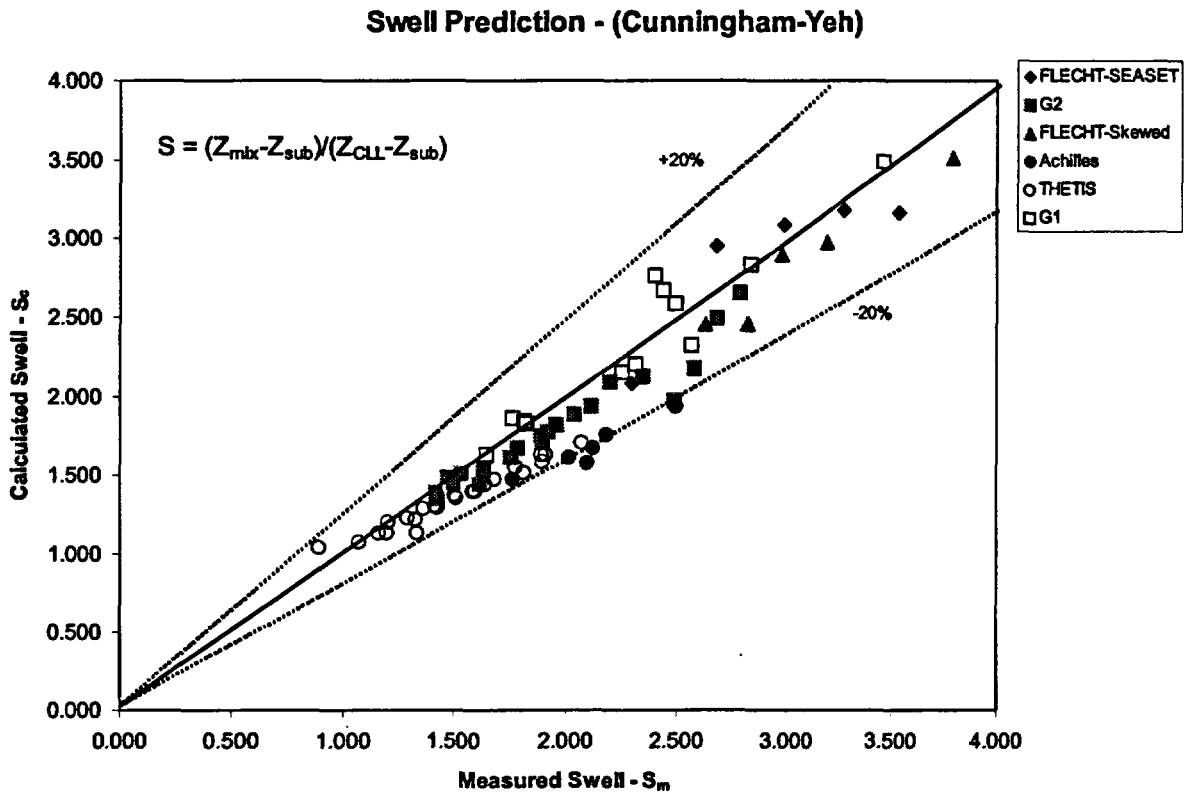


Figure G-1 Calculated Versus Predicted Swell

The comparison shows a good agreement between the Cunningham-Yeh model and the test data. Most of the data is captured within a ±20-percent band. This result provides confidence that, for a given vessel mass inventory, the core average void fraction predicted by NOTRUMP during the ADS-4/IRWST transition period is acceptable.

References

G1. J. Cunningham and H. C. Yeh, "Experiments and void Correlation for PWR Small Break LOCA Conditions," Transaction American Nuclear Society, 17, Page 369, 1973.

APPENDIX H

POST-ACCIDENT BORON CONCENTRATION ANALYSIS

This appendix is an evaluation of the potential buildup of boron in the core following cold leg loss-of-coolant accidents (LOCAs). Consideration is given to the range of break sizes from very small to double-ended cold leg breaks. The potential for boron buildup is evaluated in the early stages (before automatic depressurization system [ADS] actuation) as well as in the long term (during containment recirculation).

H1.0 SHORT-TERM EVALUATION

H1.1 SHORT-TERM ACCIDENT DEFINITION

Prior to ADS actuation, it is not likely for boron to build up significantly in the core. Normally, water circulation mixes boron in the reactor coolant system (RCS) and prevents build up in the core. In order for boron to start to build up in the core region, water circulation through the steam generators and passive residual heat removal heat exchanger (PRHR HX) has to stop. In addition, there needs to be significant injection of borated water from the core makeup tanks (CMTs) and the chemical and volume control system (CVS). For this situation to happen, the hot legs need to void sufficiently to allow the steam generator tubes to drain. Once the steam generator tubes void, the cold legs will also void since they are located higher than the hot legs. When the top of the cold legs void, the CMTs will begin to drain. When the CMTs drain to the ADS stage 1 setpoint, ADS is actuated.

H1.2 SHORT-TERM ACCIDENT EVALUATION

As shown in DCD subsection 15.6.5.4B.3.4, a 2-inch LOCA requires less than 16 minutes from the time that the hot legs void significantly until ADS is actuated. For larger LOCAs, this time difference is shorter, as seen for the 10-inch cold leg LOCA (DCD subsection 15.6.5.4B.3.6). The core boron concentration will not build up significantly in this short time. If the break is smaller than 2 inches, voiding of the hot legs will occur at a later time. With maximum operation of CVS makeup, it takes hours for the core boron concentration to build up significantly. In addition, the volume of the boric acid tank limits the maximum buildup of boron in the core.

Following a small LOCA where ADS is not actuated, the operators are guided to sample the RCS boron concentration and to initiate a post-LOCA cooldown and depressurization. The cooldown and depressurization of the RCS reduces the leak rate and facilitates recovery of the pressurizer level. Recovery of the pressurizer level allows for re-establishment of water flow through the RCS loops, which mixes the boron. The operators are also guided to take RCS boron samples during the plant cooldown. The purpose of the boron samples is to assess that there is adequate shutdown margin.

An evaluation has been performed to bound the potential core boron buildup prior to ADS actuation. Although it is thought that the progression between voiding of the hot leg and ADS actuation will not increase much as the LOCA becomes smaller and smaller, the following evaluation assumes that it may be possible for the plant to remain in this condition indefinitely. The analysis makes the following bounding assumptions:

- Appendix K decay heat.
- Maximum RCS, passive core cooling system (PXS), and CVS boric acid tank (BAT) boron concentrations.

- The RCS hot leg voids 1 hour after the accident. During this hour, the CMT recirculates/mixes with the RCS and increases its boron concentration.

- The CVS BAT injects at its maximum flow rate from time zero. Its maximum flow rate is limited by a cavitating venturi to 175 gpm.

- The CVS BAT is assumed to be completely full.

H1.3 SHORT-TERM ACCIDENT RESULTS

With these assumptions, the maximum RCS boron concentration, and the associated boron solubility temperature versus time, is calculated to be:

0.0 hr	1,200 ppm	32°F
1.0	2,625	32
2.0	9,148	76
3.0	15,510	112
4.0	21,754	137
5.0	27,535	157
6.0	33,232	175
7.0	38,917	189
8.0	39,205	190
9.0	39,205	190
10.0	39,205	190

Note that after 7 hours, the BAT empties and the core boron concentration does not increase any more. Prior to ADS actuation or operator actions to cool down the plant, the water temperature in the core will be well above these temperatures. However, ADS actuation or operator actions to cool down the RCS would introduce colder water temperatures to the core. The more likely one is where the operators initiate manual shutdown operations to cool and depressurize the plant. If the plant has not had automatic ADS for several hours after a small LOCA, the LOCA must be very small and it is most likely that the operators will be able to cool down and depressurize the RCS and avoid the need for ADS. The less likely scenario is where the operators fail to shut down the plant using normal systems and ADS is actuated several hours into the event.

During a normal cooldown, the steam generator would be used to remove the sensible RCS heat to cool down the RCS. The water entering the core would be cold leg water that is about the same temperature as the steam generator. The cold leg temperature would be close to the steam generator temperature because of the large steam generator surface area. During a natural circulation cooldown, the cold leg will be about 100°F less than the hot leg. Assuming that the steam generators are used to cool the RCS down to normal residual heat removal system (RNS) cut-in conditions (~350°F), the minimum cold leg temperature would be approximately 250°F. At these lower temperatures, the RCS pressure would be reduced, which would reduce the break flow and most likely allow a pressurizer level to be recovered. With a pressurizer level, the RNS would be aligned for shutdown cooling. RNS shutdown cooling operation would mix up the RCS boron concentration.

If ADS were actuated in such a scenario, then injection would enter the reactor vessel direct vessel injection (DVI) lines from the PXS accumulators, CMTs, and in-containment refueling water storage tank (IRWST). Because of the extended time assumed at hot conditions, the CMTs and IRWST would be heated up. The CMTs would be heated to cold leg temperatures, which are well above the solubility temperatures in the core. The IRWST would be heated up close to boiling, which is also well above the solubility temperatures in the core. The accumulator water would be the greatest challenge to the core solubility temperature because it would not be heated. In addition, its injection rate is high during ADS operation. However, there are several effects that will delay/heatup the cold water before it can enter the core. These effects include the following:

- When the accumulator injection enters the downcomer, it will spray through steam since the downcomer is not full at this time. The injection will condense some steam, which will raise the temperature of the water.

- Next, the accumulator injection water enters the water in the reactor vessel downcomer. The water in the downcomer is very hot at this time (~400°F). Initially, the cold accumulator water will be heated up by mixing with this hot water.

- Finally, as the accumulator water cools down the water in the downcomer, heat will be transferred from the hot metal of the reactor vessel, core barrel, and lower internals.

Another effect is that once ADS-4 is actuated, water will start being removed from the core as carryover out the hot leg ADS-4 path. This water flow will quickly reduce the boron concentration in the core. With an ADS-4 vent quality of 75 percent, the core boron concentration will come down to 11,800 ppm, which has a solubility temperature of 90°F. Once IRWST injection has begun, the ADS-4 vent quality is expected to be less than 60 percent, which results in a core boron concentration of 7500 ppm and a solubility temperature of 59°F.

The AP1000 Emergency Response Guidelines (ERGs) direct the operators to cool down the RCS and reduce its pressure so that the pressurizer level can be recovered and RNS shutdown cooling can be established. RNS operation will effectively mix up the RCS boron concentration. Prior to initiating RCS cooldown operations, the operators are required to sample the RCS to ensure that there is sufficient boron to provide adequate shutdown margin. Based on the concern about boron buildup in a very small LOCA, the operators will also be required to use this RCS sample to verify that the RCS boron has not built up significantly above what would be expected. Steps would be taken to limit the buildup, such as changing the CVS makeup source from only boric acid to a blend of boric acid and demineralized water. The ERGs also require several RCS samples during a natural circulation cooldown to ensure that the RCS boron concentration remains above minimum for shutdown margin. Evaluations of these samples for excessive boron buildup will also be performed.

It is expected that the operators will quickly proceed with these cooldown operations because delays could result in avoidable ADS actuation. The first RCS boron sample should be made in less than 1 hour. However, to provide additional margin in case of unanticipated delays, the operators will be required to sample the RCS within 3 hours and verify that the boron concentration has not increased excessively. Based on the above table, this would limit the maximum RCS boron concentration to less than

16,000 ppm and a solubility temperature of 112°F. The minimum core inlet temperature is not expected to drop below this temperature during a controlled cooldown or following ADS actuation.

H2.0 LONG-TERM EVALUATION

After ADS actuation, the core boron concentration will be determined by the ADS-4 vent quality. Section H2.1 addresses the minimum AP1000 water vented out of the RCS through the ADS-4 lines following a cold leg LOCA.

Section H2.2 addresses the time when the ADS-4 might not be able to vent the required amount of water. Minimum decay values are considered both just after a refueling operation and for a fresh core.

Section H2.3 addresses maximum core boron concentration both in the short-term, before ADS is actuated, and in the long-term, after ADS is actuated. This section also addresses the potential for boron precipitation in the core region, in the containment and in the ADS-4 lines.

H2.1 LONG-TERM LIQUID DISCHARGE

The AP1000 PXS injection is provided through a cold leg DVI line. With this injection arrangement, the boron concentration will tend to build up in the core region following a cold leg LOCA and ADS actuation. This buildup is limited by the ability of the ADS-4 valves to vent water from the RCS following their opening. This venting of water continues in the long-term during IRWST injection and containment recirculation operation. The long-term cooling (LTC) flow path is from the water in the containment, through the PXS recirculation lines into the reactor vessel via the DVI connections, with venting of steam/water out the ADS-4 lines.

The following provides an evaluation of the amount of water that will be vented through this path. Consideration is given to variations in plant parameters and assumptions that could affect the venting of water.

A simplified model has been developed to provide an estimate of the ADS-4 vent quality under specific conditions. Although the model is a steady-state model, it has been exercised at different times to provide the ADS-4 vent quality as a function of time.

The simplified model considers the plant operation during containment recirculation conditions, starting as early as several hours after a LOCA out through many days/weeks. The model calculates the PXS injection/ADS-4 vent quality that will remove core decay heat. The model is set up to calculate a conservatively high quality to minimize the ADS-4 water removal and, therefore, maximize the core boron buildup.

Simplified LTC model assumptions are as follows:

1. Energy balance is applied such that PXS injection/ADS-4 venting removes decay heat. Sensible heat inputs from the reactor metal components is assumed to be small by the time this analysis starts several hours after the LOCA.

2. Mass balance is applied such that PXS injection equals the ADS venting. Although Oregon State University (OSU) testing and WCOBRA/TRAC analysis both indicate that the flows tend to be cyclical, this calculation approach should approximate the actual performance using average flow rates.

3. ADS 1/2/3 flow is assumed to be insignificant relative to the ADS-4 flow. This assumption is considered reasonable and conservative. It is reasonable since in the long-term, the hot leg tends to be filled by the PXS, which cuts off the path to the pressurizer. In addition, it is conservative since its operation would reduce the ADS vent pressure loss, which would allow more PXS injection and result in a lower ADS-4 vent quality.

4. The downcomer is assumed to be filled up to the DVI connection. The fluid elevation heads are tracked from the containment water level down through the downcomer to the bottom of the core, and up through the core and the ADS-4 vent lines. WCOBRA/TRAC analysis indicates that this is a good assumption for the time frame of interest for this calculation (days/weeks).

5. The core exit quality is assumed to be equal to the ADS-4 vent quality with an adjustment to account for the higher pressure at the core exit. The core exit pressure is higher because of the head of steam/water in the upper plenum/hot leg. Since the core exit pressure is higher, the core exit quality will be slightly lower.

6. The region above the core up to a specified level in the hot legs is assumed to be filled with water except for the steam venting from the core. The steam is assumed to bubble up through this water. The bubble rise velocity is calculated to determine the resident time and the effective quality in this region. This assumption increases the backpressure on the PXS injection, which tends to reduce the PXS injection flow and increase the ADS-4 quality.

7. The ADS-4 line pressure drop is calculated using a two-phase flow multiplier. The multiplier is the maximum of homogeneous and Martinelli-Nelson. This multiplier is applied to the resistance of the line, which includes the f L/D for entrance/exits, straight pipe, elbows, tees, and valves (everything except velocity head and elevation head).

8. The velocity head differences are included in the calculation for the ADS-4 line. The hot leg velocity is assumed to be negligible. The velocity at the ADS-4 line discharge is taken at the ADS-4 valve minimum inside diameter. Homogeneous flow conditions are assumed.

9. Flow pressure losses are ignored in the reactor between the DVI discharge point and the ADS-4 inlet point. This is reasonable considering the relatively low flows that exist in these later times and the large flow areas in these sections, compared to the PXS injection lines and the ADS-4 vent lines.

10. The two-phase flow regime is determined at the entrance to the ADS-4 lines where they connect to the hot leg. This location is considered limiting because the orientation is vertical with upward flow and the velocities at this point are less than those in the individual ADS-4 lines. As a result, if the water can be lifted into the ADS-4 inlet, the water will be more easily pushed out the

individual, horizontal ADS-4 lines. The Taitlel-Dukler vertical flow regime map is used to determine the flow regime in the ADS-4 inlet lines.

Inputs provided to the model are as follows:

1. Containment pressure is taken from WGOTHIC analysis performed to determine the minimum containment pressure possible after a LOCA. Sensitivity studies are done to evaluate higher containment pressures. The containment water temperature is also taken from WGOTHIC inputs.

2. The PXS recirculation and ADS-4 line resistances are taken from the maximum, best estimate, and minimum values established for the AP1000. In addition, for maximum line resistance cases, "resident" debris is assumed to be transported to the recirculation screens and added to their pressure drop. For best-estimate resistance cases, the debris DP is assumed to be 50 percent of the maximum value. For minimum resistance cases, the debris DP is assumed to be zero.

3. The maximum decay heat is based on the AP1000 operating at 101-percent power and Appendix K decay heat values. Best-estimate and minimum decay heat values are also evaluated. Best-estimate decay heat is based on operation at 100-percent power and ANS'79 plus 0 sigma margin. Minimum decay heat is based on 2/3 of a core having operated at 100-percent power and shutdown for 20 days (for refueling) and ANS'79 minus 2 sigma margin.

4. The containment water level is determined by hand calculation. This calculation applies margins/conservatisms to the containment volumes and the water supply volumes as well as the break location. A DVI break in a PXS room results in more of the containment flooding and a lower containment water level. In addition, in the long-term (days/weeks), the containment level decreases as rooms (PXS rooms and CVS) that did not initially flood, flood due to leakage.

This model is run in two different ways. One is to assume 100-percent steam flow out the ADS-4, and the other is to vary the ADS-4 quality until the pressure drop through the system equals the driving pressure available from the containment water level.

The first analysis method is used to demonstrate that with only steam being vented and with a high hot leg level, there is margin to inject more water into the RCS. This analysis demonstrates that water will always be vented out the ADS-4. Water exiting the ADS-4 provides margin in core cooling and also limits boron buildup in the core.

Table H-1 shows the results of this analysis at two different times/conditions. One condition represents the start of recirculation for a double-ended DVI LOCA and the other one is for a non-DVI LOCA. The DVI LOCA case (1a/1b) is more limiting because it starts sooner with a lower level. Even with this case, the PXS can provide an additional 94-percent recirculation flow assuming the ADS-4 flow and quality remain the same. So even with a 75-percent hot leg level, the PXS would be filling up the hot leg further. This analysis provides strong evidence that the PXS will fill the RCS until water also leaves through the ADS-4 lines.

The second analysis method is used to evaluate the amount of water carryover out the ADS-4 lines. This analysis is performed with a range of inputs to bound the vent quality. Table H-2 illustrates the range of

conditions. Note that the two main drivers are the decay heat and the PXS flow capability. The PXS flow capability is made up of the PXS line resistances, limiting single failure of PXS valves, containment water level, and temperature.

The calculation method is iterative. An ADS-4 vent quality is guessed. The PXS injection flow is then calculated based on the energy balance. The pressure drop through the system is calculated. The ADS-4 discharge pressure is compared with the containment pressure. If the ADS-4 discharge pressure is higher, then the ADS-4 vent quality is reduced. This results in an increase in the injection/vent flows, an increase in the pressure losses in the lines, and a decrease in the ADS-4 discharge pressure. The difference between these pressures is shown in Table H-2 as the "Calc Error (ADS-4 pres – cont pr)" and is small. In addition, this result was compared with the WCOBRA/TRAC LTC analysis performed at this same time (14 days). The WCOBRA/TRAC analysis was also performed with maximum decay heat and maximum PXS line resistances. The ADS-4 vent quality calculated in WCOBRA/TRAC is slightly less than the result from the simplified hand calculation.

Table H-2 demonstrates that maximum decay heat and minimum PXS flow capability result in the highest ADS-4 vent qualities.

The calculated maximum ADS-4 vent quality as a function of time is shown below.

Time (days)	Maximum ADS-4 Quality (%)	Calculation Source
0.0	41.0	WCOBRA/TRAC
0.12	50.0	WCOBRA/TRAC
0.5	35.6	Simplified Hand Calc
1.0	27.7	Simplified Hand Calc
3.0	19.1	Simplified Hand Calc
7.0	14.4	Simplified Hand Calc
14.0	11.0	Simplified Hand Calc
30.0	8.1	Simplified Hand Calc

H2.2 DURATION OF LIQUID DISCHARGE

A simplified, bounding analysis has been performed to establish a conservative operating time for the post-LOCA LTC recirculation mode. A bounding model has been constructed that takes credit only for the buoyant effect of the steam as it bubbles up through water in the upper plenum and ADS-4 lines. Other water removal mechanisms that would exist with higher steam flows in annular, churn, slug flow regimes are ignored.

This model incorporates the following conservative assumptions:

1. The core region is assumed to contain no steam voids. Steam voids increase the driving head through the reactor and allow for LTC operation with lower decay heat/longer times.

2. The voiding in the upper plenum and ADS-4 lines is calculated assuming still water pools with no water flow (that is, ADS-4 vent quality = 100 percent). The void in these pools is calculated based on the bubble rise velocity and volumetric flow.

3. The volumetric flow of steam is based on the decay heat levels and the containment pressure and water temperature.

4. The containment pressure is assumed to be 14.7 psia.

5. The maximum containment water temperature is assumed to be 192°F based on WGOTHIC analysis.

6. The containment water level is assumed to be at minimum values.

The following equation (equation 9.34, from Wallis, G. B., "One-Dimensional Two-Phase Flow," McGraw-Hill Book Company, 1969) is used to calculate the bubble rise velocity. Note that since the void is a term in the bubble rise velocity equation, the calculation is an iterative one.

$$V = \{ 1.53 * [((st * g * (pl - pg)) / pl^{2}]^{1/4} \} / (1 - a)$$

where:

V	bubble rise velocity, ft/sec
st	water surface tension, lb/ft
g	gravitational constant, 32.2 ft/sec^2
pl	density of water, lb/ft^3
pg	density of steam, lb/ft^3
a	void in water

In Table H-3, the "Time after shutdown" is the earliest time when water may no longer be removed from the RCS through the ADS-4; at this time, steam bubbles in the upper plenum and ADS-4 volumes may not create sufficient void to raise the ADS-4 water level up to the ADS-4 discharge elevation. This time was determined using an iterative method; the time is changed until the head available to inject water from the containment balances the backpressure head caused by the steam/water mixtures in the upper plenum and ADS-4. At later times, there is less steam, which leads to less voiding and more backpressure head.

All of the five cases in Table H-3 use the same inputs/assumptions, except for the containment water level, which is varied from the minimum wall-to-wall level (103.3 ft) up to a maximum (110 ft). As

expected, higher containment water levels result in longer LTC times. This table uses a conservative minimum safety decay heat level, which is based on a condition just after refueling with 33-percent fresh fuel before the plant has returned to power.

Table H-4 shows the LTC limiting time with best-estimate decay heat instead of the minimum decay heat used in Table H-3. For these cases, the decay heat is based on ANS '79 with no margin added. The LTC times are longer because of the larger amount of decay heat.

Table H-3, Case 4 shows that with the water level at 109 feet, the LTC operation is viable for more than 6 years. Even with the minimum wall-to-wall containment water level (103.3 ft), more than 80 days of LTC operation are possible following a LOCA (Case 1). Operator actions are expected to prevent the containment water level from decreasing to the minimum wall-to-wall level and to maintain/increase that level to more than 109 feet in a week or two.

As a result, with reasonable operator actions, the LTC mode of operation is viable for more that 6 years (Table H-3, Case 4). These times after shutdown are based on the conservative assumptions used in this model. Simply by using a more reasonable decay level (that is, best-estimate decay heat), the LTC times are increased significantly. In addition, most of the cases in Tables H-3 and H-4 show steam flows (J_g) that are high enough to increase the water carryover above the values calculated in this conservative model.

The operators have the IRWST and the containment water level sensors that can measure these levels post-accident (refer to DCD Table 7.5-1). There are four redundant IRWST level sensors and three containment water level sensors provided in the plant. The IRWST sensors can measure the full tank volume. The containment water level sensors measure the containment level from the floor under the reactor vessel (elevation 71′ 6″) up to the maximum containment water level (elevation 110′). DCD Table 7.5-1 shows the range of these sensors from 72 feet up to 110 feet. In addition, the operators have the ability to sample the RCS boron concentration in a post-LOCA LTC mode. The primary sampling system (PSS) is described in DCD subsection 9.3.3. The PSS has two hot leg sample points, which can be used to take RCS water samples post-accident. The RCS connections are from the bottom of the hot legs. The system has an eductor that would be used when the RCS and containment pressures are low (DCD subsection 9.3.3.2.1).

These LTC times are based on no water carryover; some water carryover is needed to limit the boron concentration buildup. The water required to limit the concentration buildup is not large; a quality of 91-percent limits the boron concentration to 35,000 ppm. Qualities in the 60-percent range limit the boron concentration to less than 7,500 ppm, which is much less than any concentration limit. The volumetric water flows needed to reduce the quality to these levels are insignificant relative to the steam volumetric flow because of the large density difference at these conditions (density ratio = 26 lb/ft^3 / 0.0167 lb/ft^3 = 1550). ADS-4 vent qualities in the 60-percent range will not significantly affect the LTC times shown in Tables H-3 and H-4. Both WCOBRA/TRAC and the simplified model predict that the ADS-4 exit qualities will actually be low (<10 percent) in these longer times.

The decay heat from the initial core has been analyzed for short irradiation times. For such a situation, the long-term decay heat decreases more rapidly than for an equilibrium core as shown in Figure H-1.

The probability of having a LOCA early in the first core life is low. The following shows the probability of having a small or medium LOCA (with an assumed LOCA frequency of 1 E-3/yr) during the first year as well as during the whole plant life (60 years):

Irradiation Time	Prob. in 1st Year	Avg. Prob. in Plant Life
1 day	2.7 E-6	4.6 E-8/yr
7 days	1.9 E-5	3.2 E-7/yr
14 days	3.8 E-5	6.4 E-7/yr
30 days	8.2 E-5	1.4 E-6/yr

Considering a 1-week irradiation time, the probability of having a LOCA during this time with the first core is very low. Therefore, it is reasonable to analyze such a low probability event using best-estimate assumptions. The best-estimate containment water level is greater than 107 feet for at least 3 days for a DVI LOCA with no injection of the BAT or the RNS. For a non-DVI LOCA, the best-estimate level is greater than 108 feet for at least three days. Three days is more than enough time for the operators to add water to the containment to maintain and increase the water level. A minimum containment water level of 107 feet is used in this analysis.

Table H-3 (Case 2) and Table H-4 (Case 7) show that the minimum decay heat required with a containment level of 107 feet is 433.2 Btu/s. From Figure H-1, the time when this decay heat would be reached for a first core irradiated for 7 days would be more than 30 days after shutdown. This will provide the operators more than enough time to take actions to maintain and increase the containment water level. Higher water levels provide additional LTC times for a first core irritated for 7 days.

Cont Level (ft)	DH (Btu/s)	LTC Time (Days)
107	433.2	33
108	294.8	45
109	186.9	67
110	103.4	107

In conclusion, this conservative, bounding analysis shows that the AP1000 provides viable post-LOCA LTC operation for a long time (> 6 years) such that excessive boron concentration buildup will not occur.

H2.3 CORE BORON CONCENTRATION BUILDUP

The purpose of this evaluation is to calculate the maximum boron concentration buildup in the AP1000 core following a cold leg LOCA. Water leaving the RCS through ADS-4 limits the buildup of boron in the core following a cold leg LOCA. The minimum amount of water occurs with the maximum ADS-4

vent quality, which is based on the results of Section H2.1. The conservative values used in the boron concentration calculation are as shown below.

Time (days)	Maximum ADS-4 Quality (%)
0.0	60
0.12	60
0.5	45
1.0	35
3.0	25
7.0	20
14.0	15
30.0	10

The analysis of the boron concentration in the core is based on the following assumptions:

1. The core is assumed to mix only with the water in the upper plenum. A minimum mass of 27,490 pounds is used in this analysis. This mass is based on WCOBRA/TRAC analysis.

2. The only boron that leaves the core is that carried out of the RCS in water in the ADS-4 vent flow. Because of the low volatility of boron, little boron will be present in the vapor phase. Based on test results (Byrnes, D. E., "Some Physicochemical Studies of Boric Acid Solutions at High Temperatures," WCAP-3713 (Proprietary), September 1962), the amount of boron present in the vapor phase is only about 1 percent of the concentration of boron in the water. This data is applicable to water that is partially neutralized with lithium hydroxide (typical PWR practice). For calculating the maximum core boron concentration, the volatility of boron is ignored.

3. The maximum initial post-accident boron concentration is calculated based on the following:

   - RCS – minimum volume, maximum boron concentration
   - CMTs, accumulator, IRWST – maximum volume and boron concentration
   - BATs, cask loading pit – maximum volume and boron concentration

   The minimum RCS volume is used because its maximum concentration is less than the concentration after mixing with the other volumes. The BATs (both) are assumed to be injected by the CVS makeup pumps, and the cask loading pit is assumed to be injected by the RNS pumps. Although neither the CVS nor the RNS pumps are safety-related, their operation is likely and it makes the initial mixed boron concentration higher.

The core concentration starts at about 2980 ppm and reaches a maximum value of 7400 ppm in about 2.4 hours. Figure H-2 shows the calculated AP1000 core boron concentration as a function of time. This figure also shows the solubility temperature of the boron solution in the core. When the core is at its

maximum concentration of 7400 ppm, the solubility temperature is 58°F. The core inlet temperature will be well above this temperature based on several considerations.

First, the IRWST is located inside containment and is expected to normally be above this temperature, although the Technical Specifications allow the IRWST to be as cold as 50°F. Second, the IRWST injection water will heat up significantly between the time it enters the RCS and the time it enters the core region, especially in this earlier time frame. IRWST flow will heat up because it sprays through steam as it drops down from the DVI injection connection to the water level in the downcomer. The IRWST water also heats up because it comes in contact with the hot metal surfaces of the reactor vessel, the core barrel, and the core support plate. The LTC WCOBRA/TRAC analysis shows that the minimum heatup of the IRWST water is 70°F (120° to 190°F) during the initial IRWST injection phase. This analysis also shows that the temperature of water entering the core gradually increases to about 240°F as the IRWST drains. This further increase is due to a reduction in the IRWST flow (as the IRWST water level decreases) and IRWST heatup due to ADS stage 1/2/3 operation. Even with only a 70°F heatup and the minimum Technical Specification IRWST temperature of 50°F, the minimum core inlet temperature would be 120°F. This temperature is comfortably above the solubility temperature of 58°F. After recirculation starts, the margin becomes even greater because of the increase in the injection water temperature and because the solubility temperature begins to decrease (as the quality decreases).

Note that later on in the IRWST injection phase, as the IRWST level decreases, the IRWST injection temperature will increase. This increase is due to ADS 1/2/3 operation and to steam condensation returned to the IRWST by the gutter. For a DVI LOCA, the ADS 1/2/3 operation admits less mass/energy into the IRWST because of the rapid sequencing of the ADS-4 valves. For a DVI LOCA, the top portion of the IRWST is calculated to heat up ~20°F. Note that ADS 1/2/3 operation is assumed only to affect the top portion of the IRWST from the bottom of the sparger up.

The boron concentration of the water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is about 2960 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

No buildup of boron is expected in the RCS hot legs and ADS-4 vent paths because these areas always see a flow of hot water and steam. Note that although the hot water contains boron, its concentration is far below the solubility limit; when the core is at its maximum boron concentration of 7400 ppm the water is capable of holding about 80,000 ppm boron (at 240°F). Even assuming that the inside surface of the ADS-4 piping is at the containment temperature of 176°F, the water could still hold 33,000 ppm boron. Boron in the steam will be at a much lower concentration (about 1 percent of that in the water, or about 74 ppm in this case). The only way that the boron in the steam could plate out would be for the steam to be condensed and then have the water evaporate. Such a process could not happen inside the ADS piping with the continued high flow of hot water.

Table H-1 Ability of PXS to Fill Hot Legs

		Case 1a	Case 1b	Case 2a	Case 2b
Basis for decay heat		Max	Max	Max	Max
for PXS Injection		Min	Min	Min	Min
LOCA		DVI	DVI	non-DVI	non-DVI
Time after accident (hr)		2.5	2.5	5.0	5.0
Total containment pres (psia)		14.7	14.7	14.7	14.7
Recirc water temp (F)		175.2	175.2	192.0	192.0
Recirc water elev. (ft)		108.0	108.0	109.3	109.3
HL mixture level		75%	75%	75%	75%
Upper plenum collapsed level		45%	45%	45%	45%
Core collapsed level		53%	54%	53%	54%
ADS 4 vent quality		100.0%	85.3%	100.0%	60.3%
PXS Flows					
Recirc flow	(lb/sec)	42.94	50.03	36.13	59.16
ADS 4 water vent	(lb/sec)	0.00	7.36	0.00	23.51
ADS 4 steam vent	(lb/sec)	42.94	42.67	36.13	35.65
ADS 4 gas vel	(m/sec)	204.58	203.29	172.13	169.85
ADS 4 flow regime		annular	annular	annular	annular
Pressures					
Containment	(psia)	14.7	14.7	14.7	14.7
ADS 4 Discharge	(psia)	16.0	14.7	18.1	14.7
Pres Diff, ADS-cont pres	(psi)	1.31	0.00	3.40	0.00
Pressure losses					
Cont Recirc line	(psi)	0.47	0.64	0.34	0.91
ADS 4 line	(psi)	4.97	6.08	3.51	6.26
ADS 4 2-phase multiplier		703	675	699	518
Increase in recirc possible(1)		94%	0%	233%	0%

Note (1) Shows the increase in PXS recirc flow that would occur to use up the available DP, assuming the ADS vent flow & quality remained unchanged.

Table H-2 Effect of Inputs of ADS-4 Vent Quality

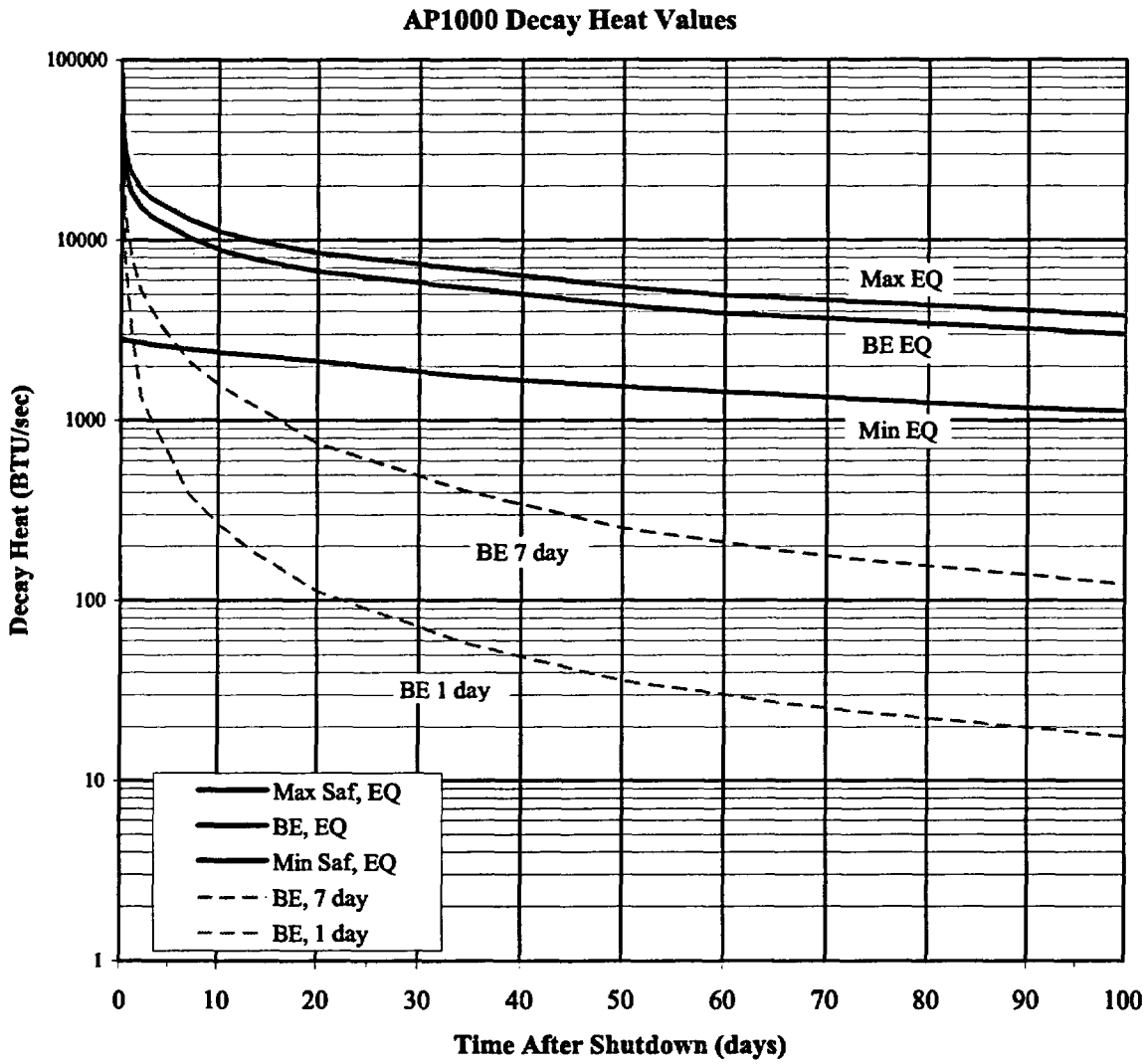
		Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8
Basis for decay heat		Max	BE	Max	Max	Max	BE	Min
for containment flood level		MinMin	MinMin	BE	MinMin	MinMin	BE	Max
for containemnt pressure		MinMin	MinMin	MinMin	BE	MinMin	BE	Max
for flow resistance		Max	Max	Max	Max	BE	BE	Min
LOCA location		DVI	DVI	DVI	DVI	DVI	DVI	DVI
Number ADS 4 valves open		3	3	3	3	4	4	4
recirc valves open		2	2	2	2	4	4	4
Time after accident (days)		14.00	14.00	14.00	14.00	14.00	14.00	14.00
Decay power (BTU/sec)		9946	7754	9946	9946	9946	7754	3421
Total containment pres (psia)		14.7	14.7	14.7	20.2	14.7	20.2	24.8
Recirc water temp (F)		190.4	190.4	190.4	190.4	187.4	187.4	185.4
Recirc water elev. (ft)		103.5	103.5	107.4	103.5	103.5	107.4	108.2
HL level (%)		75%	75%	75%	75%	75%	75%	75%
ADS 4 vent quality		8.27%	5.98%	6.87%	6.90%	3.32%	0.38%	0.05%
Core exit quality		6.90%	4.56%	5.24%	5.70%	2.14%	0.10%	0.01%
Core collapsed liquid level		69%	79%	75%	80%	87%	100%	100%
Core / top HL collapsed level		44%	51%	45%	53%	51%	85%	90%
PXS Flows								
Recirc flow	(lb/sec)	97.58	97.22	112.53	95.01	174.69	172.03	62.15
ADS 4 water vent	(lb/sec)	89.52	91.41	104.80	88.46	168.89	171.38	62.12
ADS 4 steam vent	(lb/sec)	8.07	5.81	7.73	6.56	5.80	0.65	0.03
ADS 4(A) gas vel.	(m/sec)	19.50	14.06	18.69	11.75	20.40	1.69	0.06
ADS 4(B) gas vel.	(m/sec)	38.43	27.70	36.83	23.16	21.23	1.75	0.07
ADS 4(A) flow regime		annular	slug/churn	annular	slug/churn	annular	slug/churn	bubble
ADS 4(A) flow regime		annular	annular	annular	annular	annular	slug/churn	bubble
Pressures:								
Containment	(psia)	14.70	14.70	14.70	20.24	14.70	20.24	24.79
DVI connection	(psia)	13.56	13.58	14.33	19.24	14.54	21.77	28.26
Bottom core	(psia)	21.06	21.08	21.84	26.75	22.06	29.29	35.78
Top core	(psia)	18.78	18.19	19.06	23.69	18.04	23.52	30.00
Top HL	(psia)	17.53	16.75	17.78	22.18	16.60	21.13	27.47
ADS 4 discharge	(psia)	14.72	14.70	14.70	20.24	14.70	20.24	24.79
ERROR (Cont pres-ADS 4 pres)		2.4E-02	4.8E-07	8.8E-07	8.0E-07	8.0E-08	3.5E-07	6.09E-07
Pressure losses								
Cont Recirc line	(psi)	2.46	2.45	3.28	2.34	1.52	1.47	0.14
(screen debris)	(psi)	0.32	0.32	0.37	0.31	0.29	0.28	0.00
ADS 4 line (friction)	(psi)	1.60	1.16	1.74	1.20	0.97	0.08	0.00
(velocity head)	(psi)	1.17	0.85	1.30	0.69	0.86	0.08	0.00
2-phase flow multiplier		97.7	71.3	79.9	76.5	41.4	3.6	1.0
Fluid Densities:								
Recirc water	(lb/ft3)	60.33	60.33	60.33	60.34	60.40	60.41	60.45
DVI conn.	(lb/ft3)	60.33	60.33	60.33	60.34	60.40	60.41	60.45
Fuel inlet	(lb/ft3)	60.34	60.34	60.34	60.34	60.41	60.41	60.45
Avg fuel	(lb/ft3)	23.50	29.78	28.51	31.42	41.29	59.26	59.47
Fuel to top HL	(lb/ft3)	28.89	33.32	29.71	34.88	33.27	55.35	58.65
ADS 4	(lb/ft3)	0.45	0.62	0.54	0.72	1.10	10.95	40.67
ADS 4 void		99%	99%	99%	99%	98%	82%	31%

Table H-3 AP1000 Limiting Long-Term Cooling Conditions With Minimum Decay Heat

	Case 1	Case 2	Case 3	Case 4	Case 5
Time after shutdown (days)	88.3	823.2	1,313.1	2,497.6	7,069.2
, (years)	0.2	2.3	3.6	6.8	19.4
Decay heat basis	Min	Min	Min	Min	Min
Decay heat, BTU/sec	1776.7	433.2	294.8	186.9	103.4
Cont pres (psia)	14.7	14.7	14.7	14.7	14.7
Cont water temp (F)	192.0	192.0	192.0	192.0	192.0
Water subcooling (F)	20.0	20.0	20.0	20.0	20.0
Cont water level, ft	103.3	107.0	108.0	109.0	110.0
ADS 4 vent quality	100%	100%	100%	100%	100%
Core steam voiding	0%	0%	0%	0%	0%
Flows, Total (lb/sec)	1.794	0.437	0.298	0.189	0.104
, Steam (lb/sec)	1.794	0.437	0.298	0.189	0.104
, Water (lb/sec)	0.000	0.000	0.000	0.000	0.000
Void, upper core	0.0%	0.0%	0.0%	0.0%	0.0%
, upper plenum	19.7%	5.3%	3.6%	2.3%	1.2%
, ADS 4	79.2%	47.3%	37.5%	27.4%	17.2%
Bubble rise, core (ft/sec)	na	na	na	na	na
, upper plenum (ft/sec)	2.31	1.95	1.92	1.89	1.87
, ADS 4 pipe (ft/sec)	8.93	3.53	3.00	2.57	2.25
Jg (gas vel), m/sec	6.27	1.48	1.00	0.63	0.34
Jl (liq vel), m/sec	0.00	0.00	0.00	0.00	0.00
Densities (lb/ft3)					
Rrecirc / downcomer	60.30	60.30	60.30	60.30	60.30
Core	59.81	59.81	59.81	59.81	59.81
Upper plenum	49.74	56.47	57.37	58.07	58.63
ADS 4 pipe	13.22	31.92	37.64	43.50	49.48
Pressures (psia)					
Containment	14.70	14.70	14.70	14.70	14.70
Bottom fuel	23.76	25.31	25.73	26.15	26.56
Top fuel	17.94	19.49	19.91	20.33	20.75
Top HL	15.57	16.80	17.17	17.56	17.95
ADS 4 discharge	14.70	14.70	14.70	14.70	14.70

Table H-4 AP1000 Limiting Long-Term Cooling Conditions With Best Estimate Decay Heat

	Case 6	Case 7	Case 8	Case 9	Case 10
Time after shutdown (days)	425.3	1,465.9	2,585.3	5,566.5	12,146.1
, (years)	1.2	4.0	7.1	15.3	33.3
Decay heat basis	BE	BE	BE	BE	BE
Decay heat, BTU/sec	1231.8	433.2	293.0	186.4	103.4
Cont pres (psia)	14.7	14.7	14.7	14.7	14.7
Cont water temp (F)	192.0	192.0	192.0	192.0	192.0
Water subcooling (F)	20.0	20.0	20.0	20.0	20.0
Cont water level, ft	104.2	102.0	108.0	109.0	110.0
ADS 4 vent quality	100%	100%	100%	100%	100%
Core steam voiding	0%	0%	0%	0%	0%
Flows, Total (lb/sec)	1.244	0.437	0.296	0.188	0.104
, Steam (lb/sec)	1.244	0.437	0.296	0.188	0.104
, Water (lb/sec)	0.000	0.000	0.000	0.000	0.000
Void, upper core	0.0%	0.0%	0.0%	0.0%	0.0%
, upper plenum	14.3%	5.3%	3.6%	2.2%	1.2%
, ADS 4	72.3%	47.3%	37.5%	27.4%	17.2%
Bubble rise, core (ft/sec)	na	na	na	na	na
, upper plenum (ft/sec)	2.16	1.95	1.92	1.89	1.87
, ADS 4 pipe (ft/sec)	6.73	3.53	2.98	2.56	2.25
Jg (gas vel), m/sec	4.32	1.48	0.99	0.62	0.34
Jl (liq vel), m/sec	0.00	0.00	0.00	0.00	0.00
Densities (lb/ft3)					
Rrecirc / downcomer	60.30	60.30	60.30	60.30	60.30
Core	59.81	59.81	59.81	59.81	59.81
Upper plenum	52.07	56.47	57.37	58.08	58.63
ADS 4 pipe	17.27	31.92	37.64	43.50	49.48
Pressures (psia)					
Containment	14.70	14.70	14.70	14.70	14.70
Bottom fuel	24.14	25.31	25.73	26.15	26.56
Top fuel	18.32	19.49	19.91	20.33	20.75
Top HL	15.83	16.80	17.17	17.56	17.95
ADS 4 discharge	14.70	14.70	14.70	14.70	14.70



AP1000 Decay Heat Values

Shutdown (days)	Equil Core 6 year Max Saf	Equil Core 6 year BE	Equil Core 6 year Min Saf	First Core 1 day BE	First Core 7 day BE	
0.417	30148	83435	65967	2806	50562	BTU/sec
1	23433	18527	2748	3367	8204	BTU/sec
7	13012	10288	2474	390	2106	BTU/sec
14	9807	7754	2283	179	1177	BTU/sec
35	6807	5382	1746	57	402	BTU/sec
90	4055	3206	1174	20	139	BTU/sec

Figure H-1 AP1000 Decay Heat Values

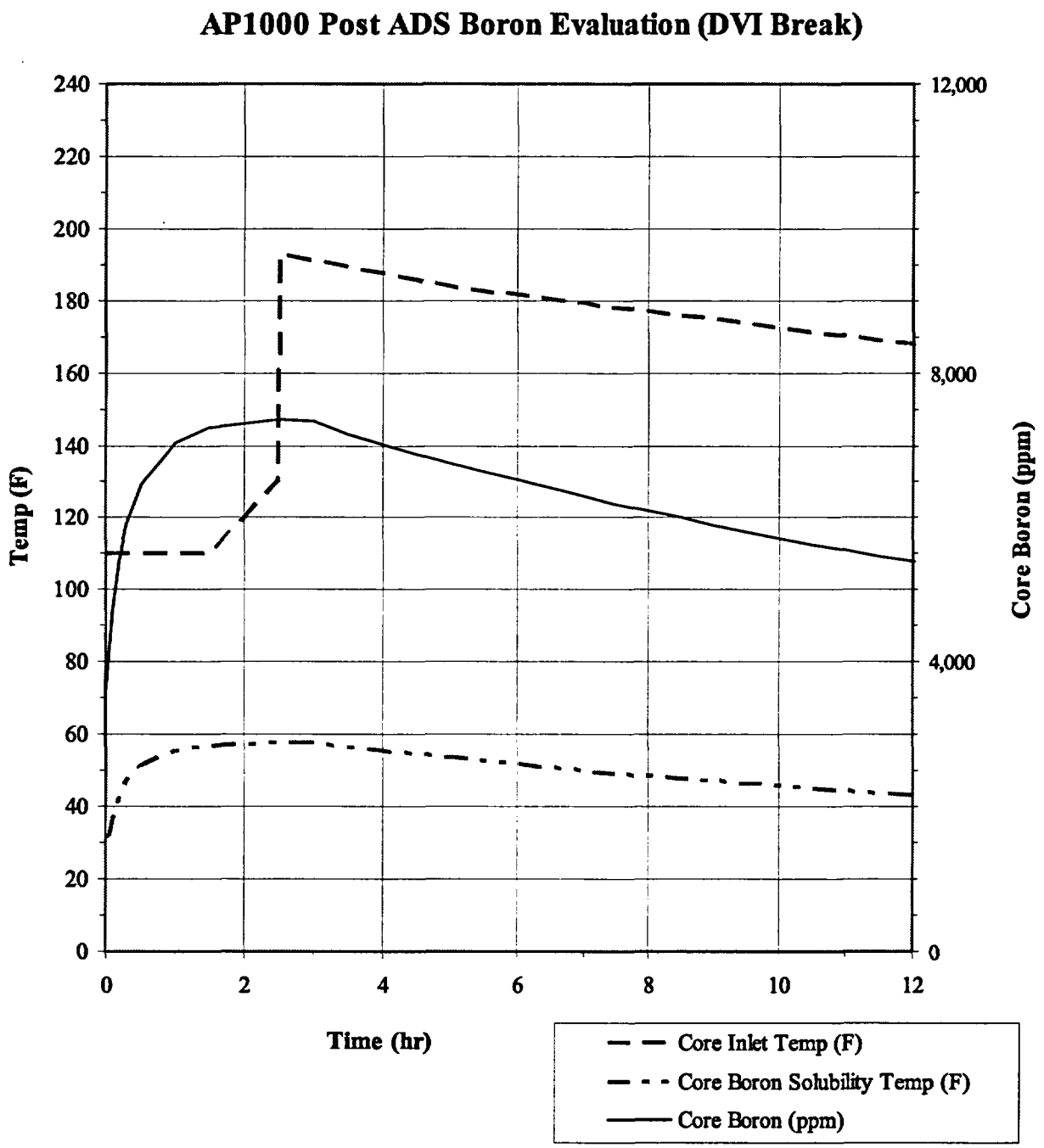


Figure H-2 Calculated AP1000 Core Boron Concentration as a Function of Time