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Your ref: Docket No. 52-006
Our ref: DCP/NRC1627

September 19, 2003

SUBJECT: Transmittal of Westinghouse Documents, "AP1000 Code Applicability Report," WCAP-15644-P, Rev. 1 (Proprietary) and WCAP-15644-NP, Rev. 1 (Non-Proprietary) dated September 2003

Attached please find proprietary and non-proprietary versions of Revision 1 of WCAP-15644, "AP1000 Code Applicability Report," dated September 2003. This report is updated consistent with our responses to open items in the Draft Safety Evaluation Report previously transmitted to the NRC.

The Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit are also enclosed with this submittal letter as Enclosure 1. Attachment 1 contains Westinghouse proprietary information consisting of trade secrets, commercial information or financial information which we consider privileged or confidential pursuant to 10 CFR 2.790. Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosures.

This material is for your internal use only and may be used for the purpose for which it is submitted. It should not be otherwise used, disclosed, duplicated, or disseminated, in whole or in part, to any other person or organization outside the Commission, the Office of Nuclear Reactor Regulation, the Office of Nuclear Regulatory Research and the necessary subcontractors that have signed a proprietary non-disclosure agreement with Westinghouse without the express written approval of Westinghouse.

D043

Correspondence with respect to the application for withholding should reference AW-03-1705, and should be addressed to Hank A. Sepp, Manager of Regulatory and Licensing Engineering, Westinghouse Electric Company, P.O. Box 355, Pittsburgh, Pennsylvania, 15230-0355.

Please contact me at 412-374-5355 if you have any questions concerning this submittal.

Very truly yours,

M. M. Corletti

Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosure

1. Westinghouse Electric Company Copyright Notice, Proprietary Information Notice, Application for Withholding, and Affidavit AW-03-1705.

/Attachments

1. WCAP-15644-P Revision 1, AP1000 Code Applicability Report, dated September 2003
2. WCAP-15644-NP Revision 1, AP1000 Code Applicability Report, dated September 2003

DCP/NRC1627
Docket No. 52-006

September 19, 2003

Enclosure 1

Westinghouse Electric Company
Application for Withholding and Affidavit

Westinghouse

Westinghouse Electric Company
Nuclear Power Plants
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

September 19, 2003

AW-03-1705

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Mr. John Segala

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Transmittal of Westinghouse Proprietary Class 2 Document WCAP-15644-P, Rev. 1, "AP1000 Code Applicability Report"

Dear Mr. Segala:

The application for withholding is submitted by Westinghouse Electric Company, LLC ("Westinghouse") pursuant to the provisions of paragraph (b)(1) of Section 2.790 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject documents. In conformance with 10 CFR Section 2.790, Affidavit AW-03-1705 accompanies this application for withholding setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.790 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-03-1705 and should be addressed to the undersigned.

Very truly yours,

M. M. Corletti
Passive Plant Projects & Development
AP600 & AP1000 Projects

/Enclosures

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AW-03-1705

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared H. A. Sepp, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company, LLC ("Westinghouse"), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief.

H. A. Sepp, Manager
Regulatory Compliance & Plant Licensing
Nuclear Power Plants Business Unit
Westinghouse Electric Company, LLC

Sworn to and subscribed
before me this 19th day
of September, 2003

Sharon L. Fiori
Notary Public

SHARON L. FIORI
COMMONWEALTH OF PENNSYLVANIA
NOTARY PUBLIC

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

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(1) I am Manager, Regulatory Compliance and Plant Licensing, in the Nuclear Power Plants Business Unit, of the Westinghouse Electric Company LLC ("Westinghouse"), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Electric Company, LLC.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by the Westinghouse Electric Company, LLC in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Attachment 2 as Proprietary Class 2 in the Westinghouse Electric Co., LLC document: (1) "AP1000 Design Certification Review, Draft Safety Evaluation Report Open Item Response."

This information is being transmitted by Westinghouse's letter and Application for Withholding Proprietary Information from Public Disclosure, being transmitted by Westinghouse Electric Company (W letter AW-03-1705) and to the Document Control Desk, Attention: John Segala, DIPM/NRLPO, MS O-4D9A.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation supporting determination of APP-GW-GL-700, "AP1000 Design Control Document," analysis on a plant specific basis

(b) Provide the applicable engineering evaluation which establishes the Tier 2 requirements as identified in APP-GW-GL-700.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for Licensing Documentation.

(b) Westinghouse can sell support and defense of AP1000 Design Certification.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar methodologies and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for performing and analyzing tests.

Further the deponent sayeth not.

DCP/NRC1627
Docket No. 52-006

September 19, 2003

Attachment 2

WCAP-15644-NP Revision 1

"AP1000 Code Applicability Report"

Westinghouse Non-Proprietary Class 3

WCAP-15644-NP
Revision 1

September 2003

AP1000 Code Applicability Report

Westinghouse

WCAP-15644-NP
Revision 1

AP1000 Code Applicability Report

September 2003

AP1000 Document: APP-GW-GSC-003, Revison 1

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

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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 to 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 over-predicted 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 are 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.

2.4 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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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 (cont.) Assessment of the AP1000 LBLOCA Processes

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

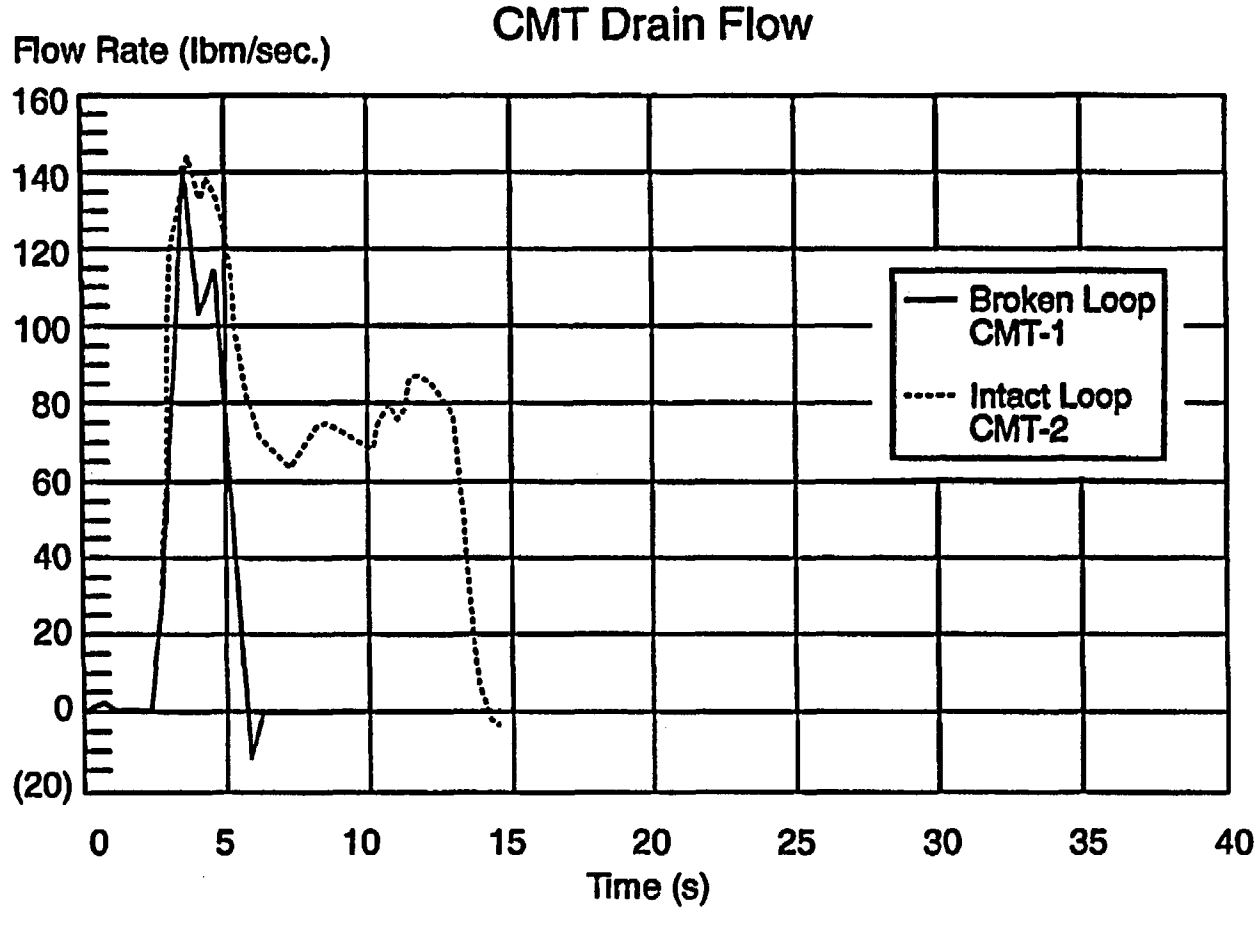


Figure 2-1 CMT Injection for an AP600 LBLOCA

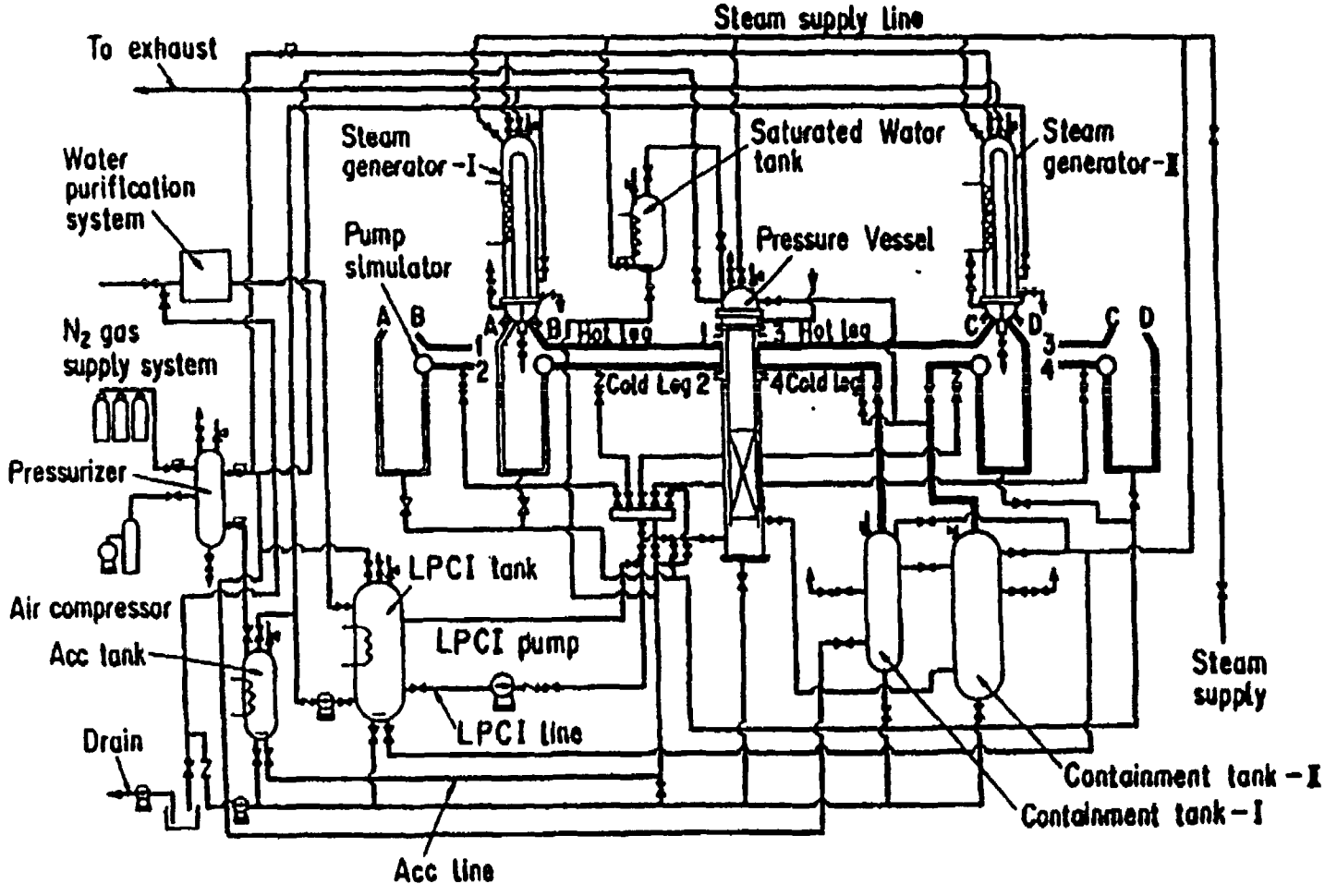


Figure 2-2 Japanese CCTF Test Facility

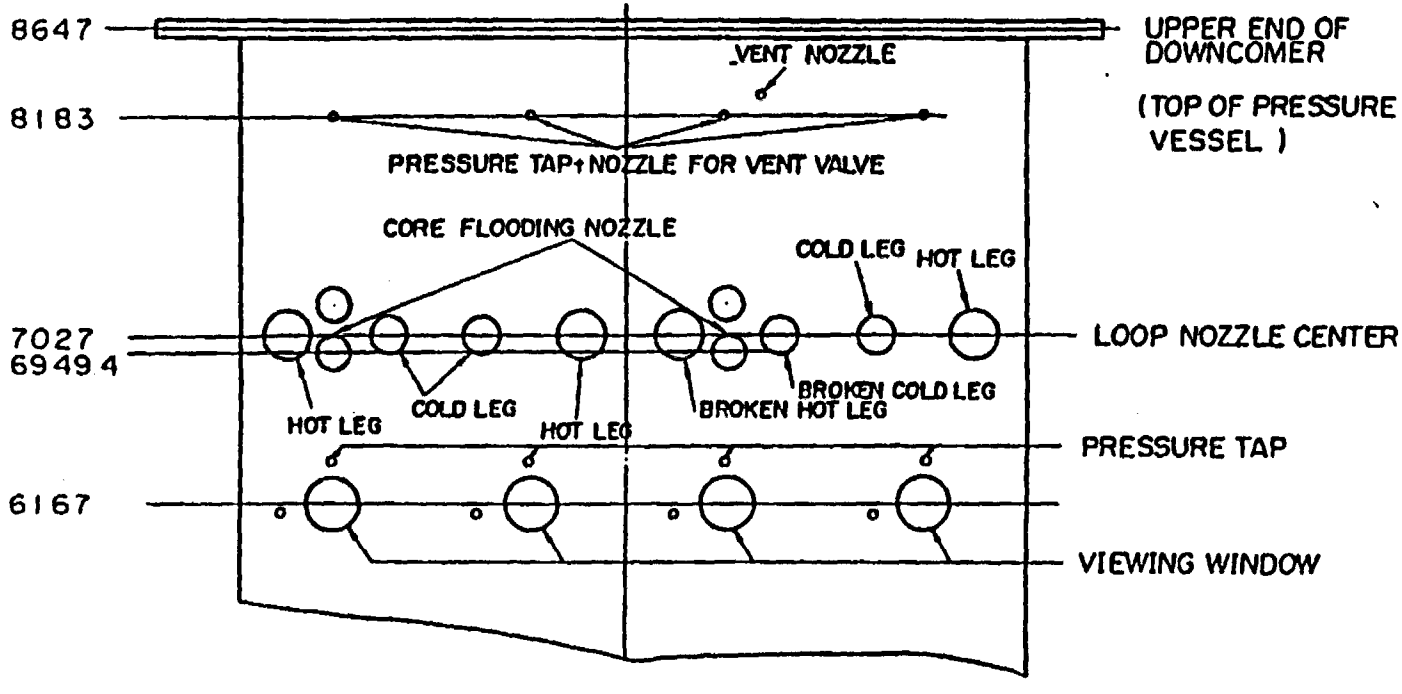
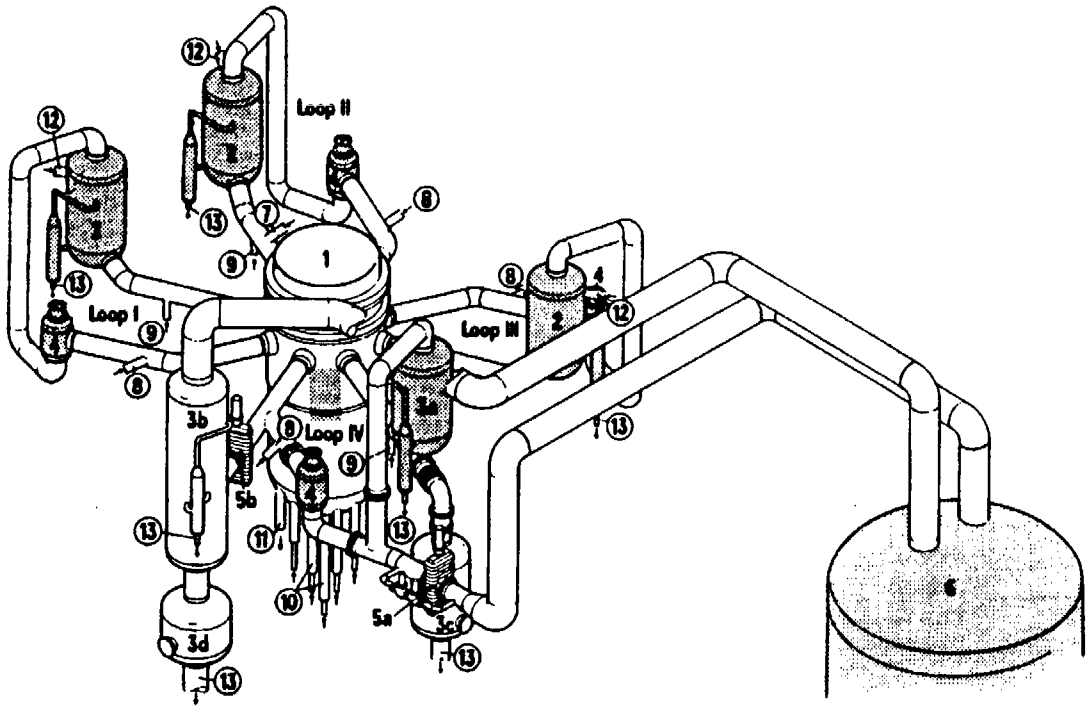


Figure 2-3 Japanese CCTF Vessel and Downcomer Geometry

SIEMENS



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

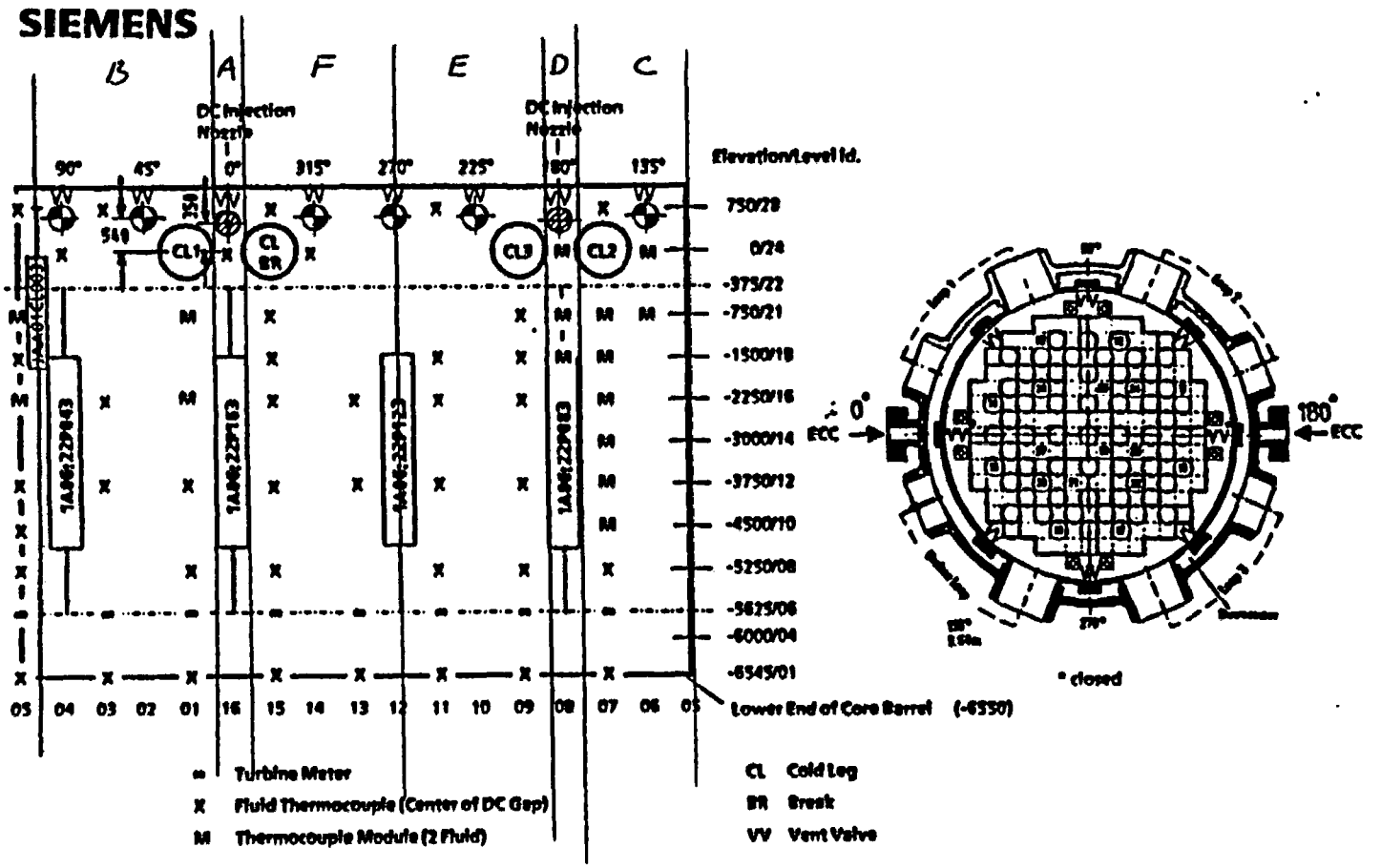


Figure 2-5 Upper Plenum Test Facility Vessel and Downcomer Geometry

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)

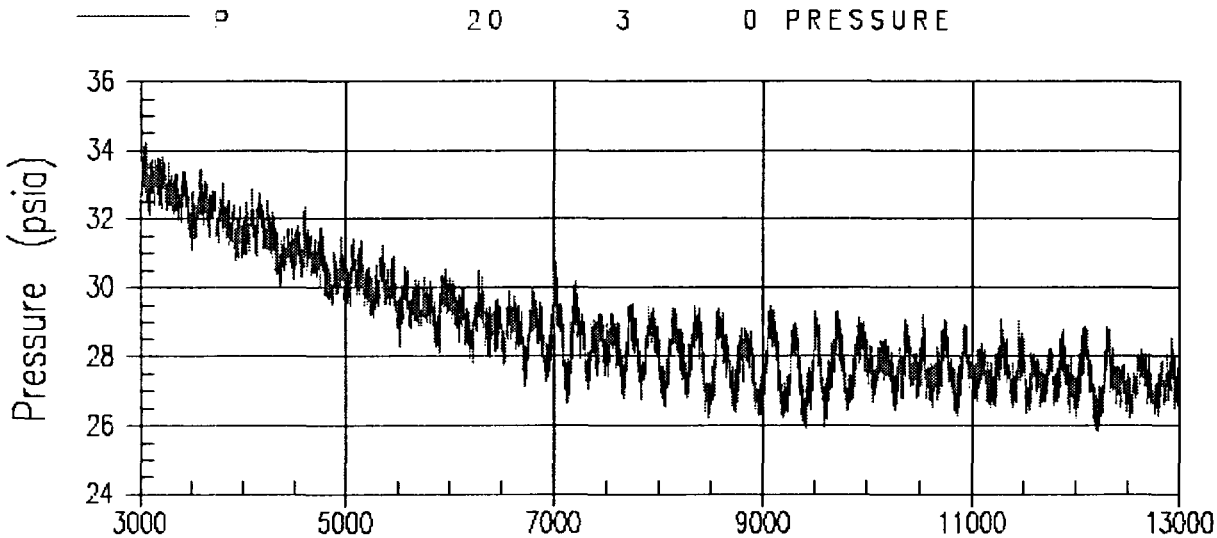


Figure 2-8 Upper Plenum Pressure

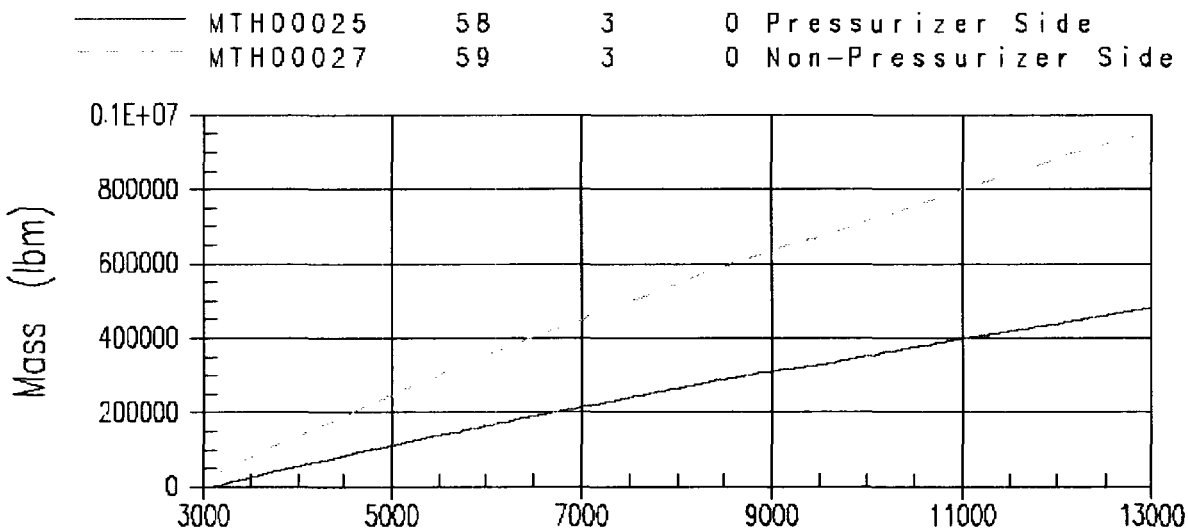


Figure 2-9 Integrated ADS4-1 and ADS4-2 Flows

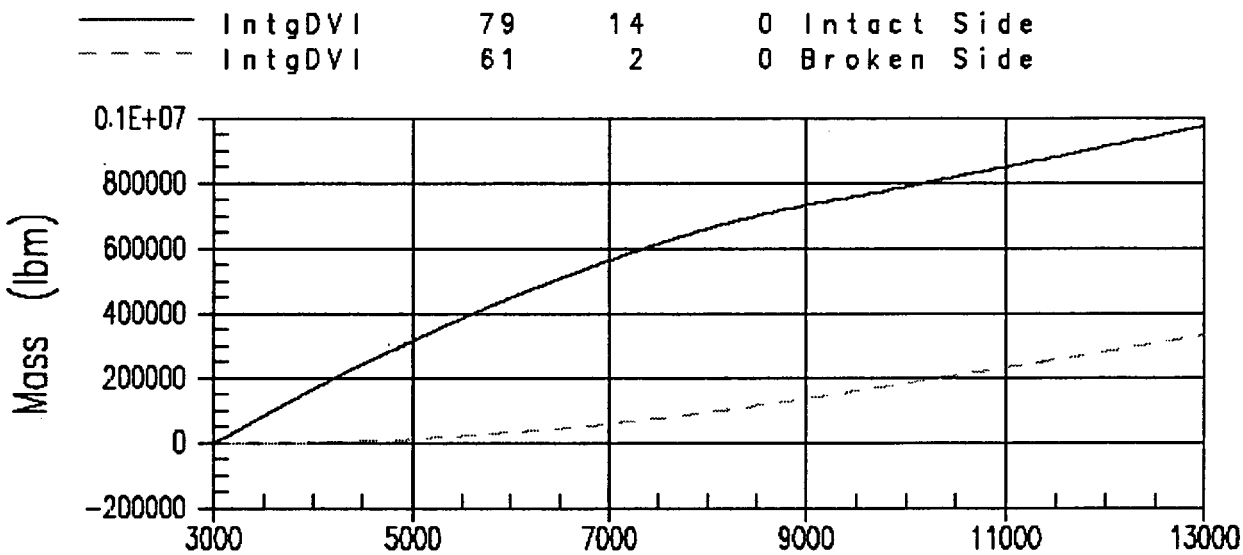


Figure 2-10 Integrated DVI Injection Flow

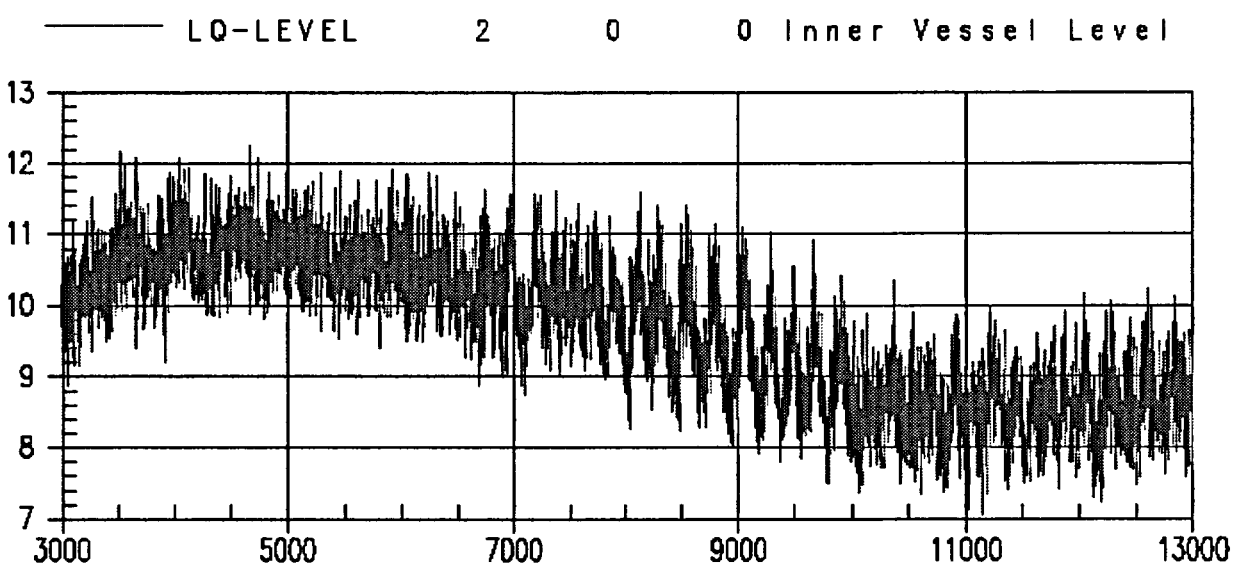


Figure 2-11 Inner Vessel Collapsed Liquid Level Above Bottom of Active Fuel

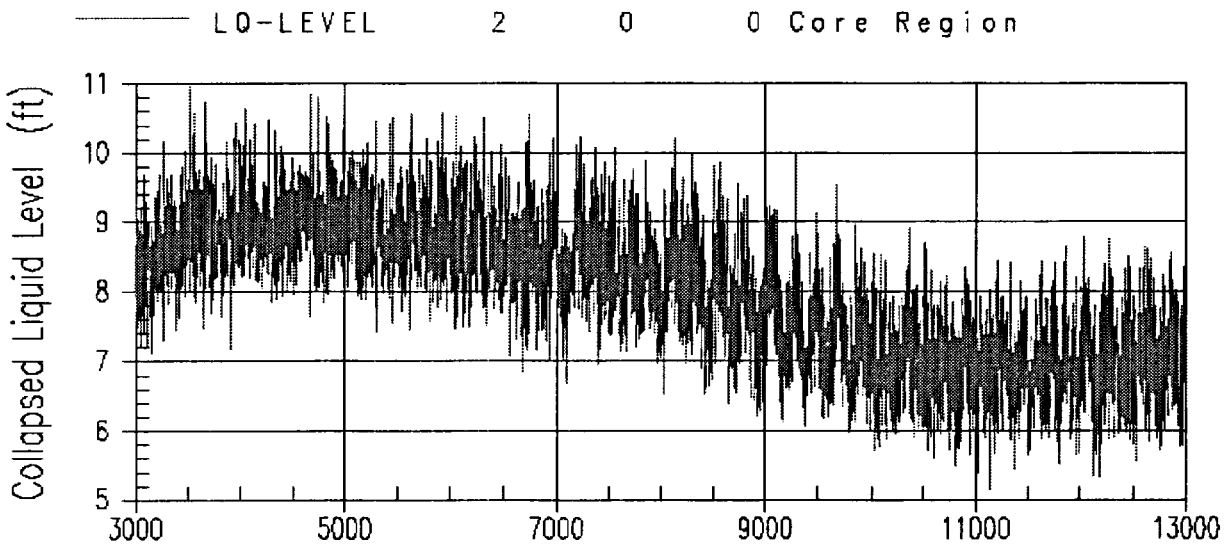


Figure 2-12 Core Collapsed Liquid Level Above Bottom of Active Fuel

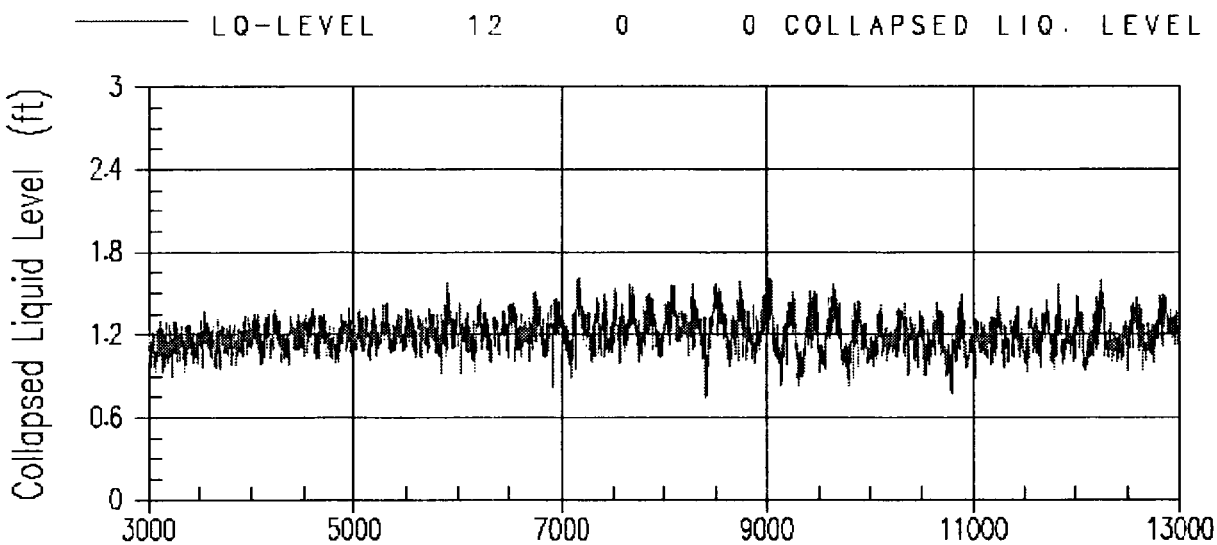


Figure 2-13 Hot Leg 2

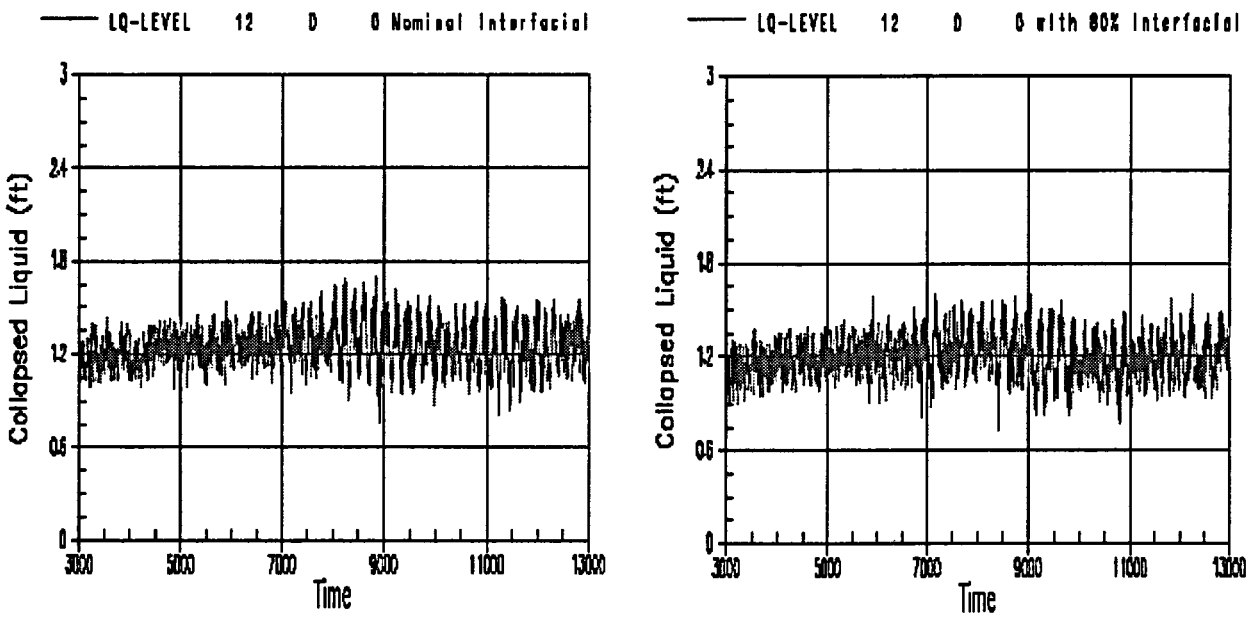


Figure 2-14 Hot Leg 2 with Nominal Interfacial Drag, and Hot Leg 2 with 80-Percent Interfacial Drag

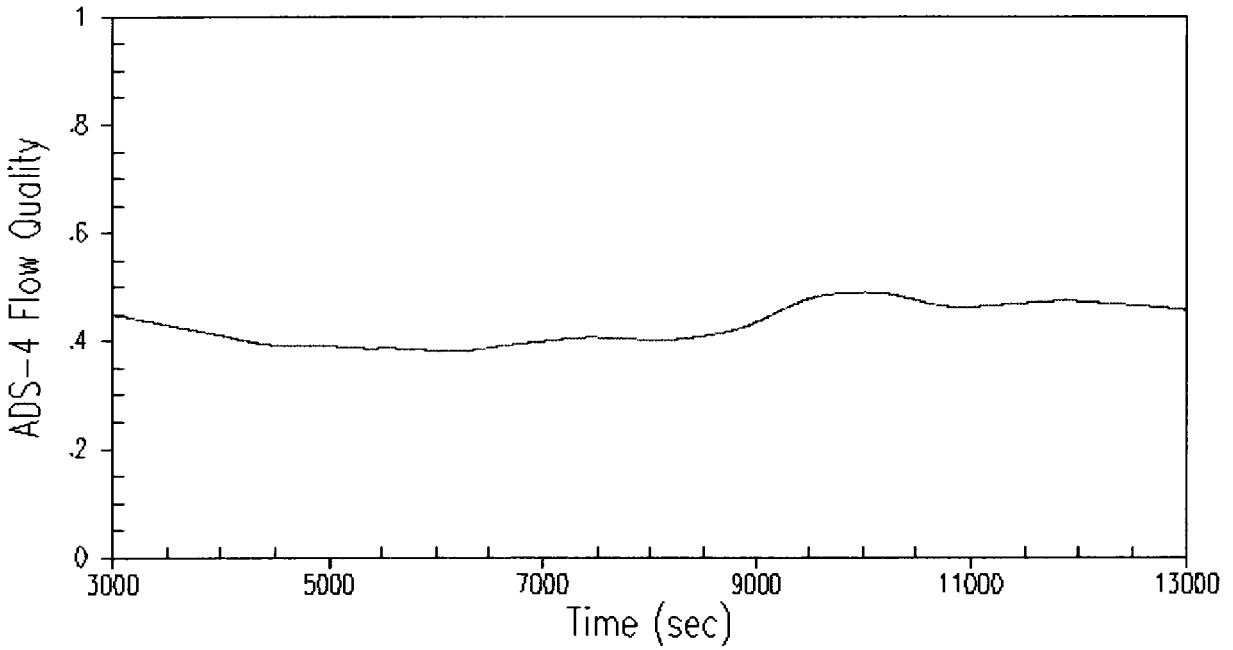


Figure 2-15 ADS-4 Exit Quality

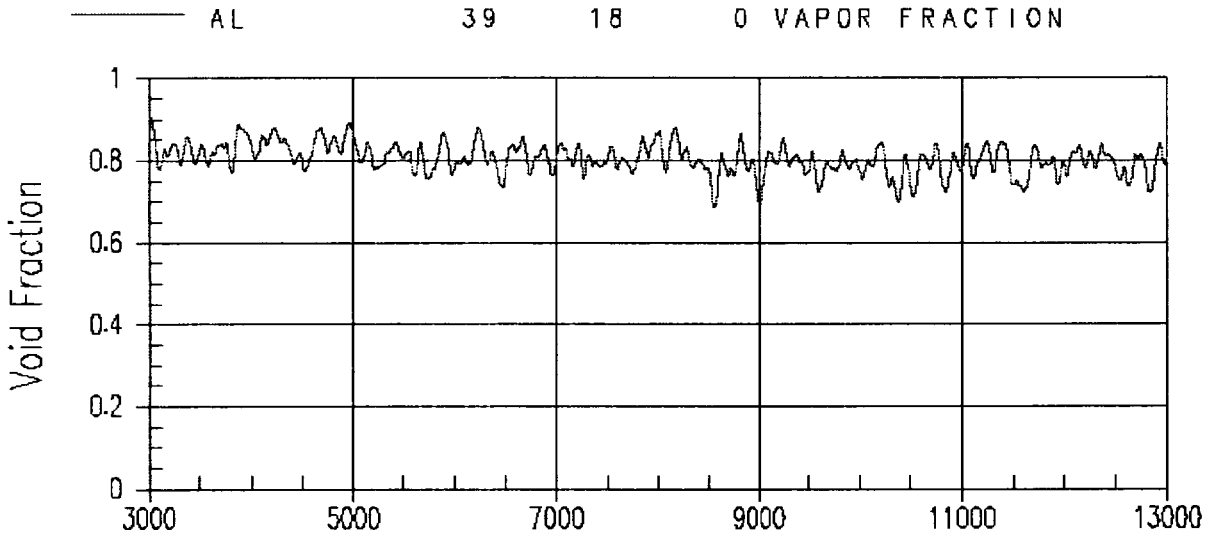


Figure 2-16 Void Fraction – TOP of Hot Assembly

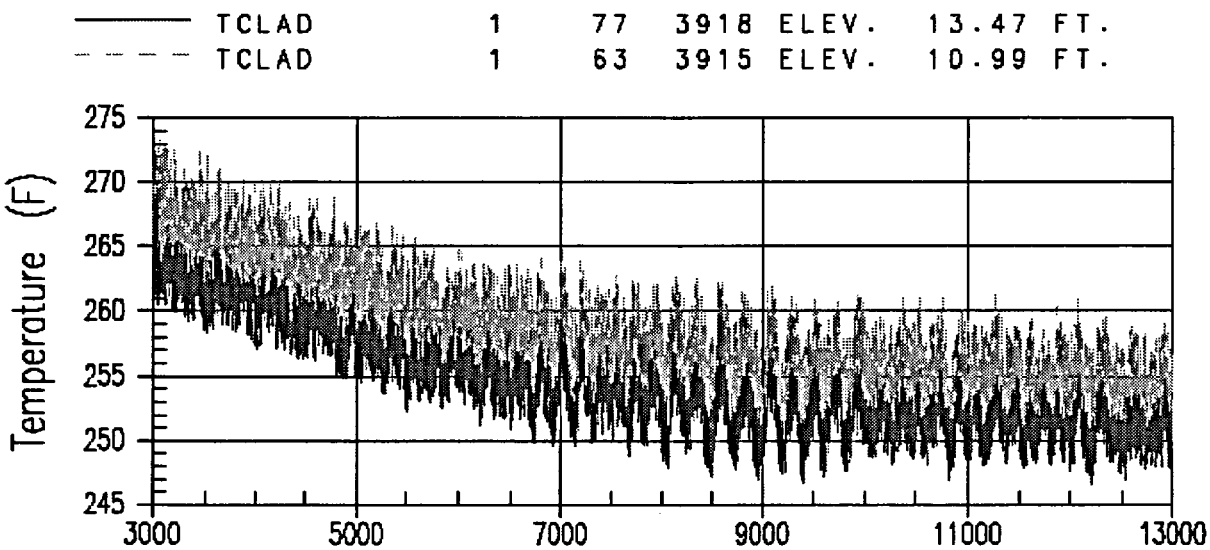


Figure 2-17 Cladding Temperatures at Higher Elevations

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 over-riding 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. The following sections present the major areas of review and the conclusions reached by the ACRS and the NRC staff.

3.2.1.1 SBLOCA Analysis Results

The SBLOCA analyses performed in support of the AP600 design met the following acceptance criteria for the calculated ECCS performance:

- *The calculated peak cladding temperature (PCT) is less than 1204°C (2200°F).*
- *The calculated total oxidation of the cladding is within 0.17 times the total cladding thickness before oxidation.*
- *The calculated total amount of hydrogen generated is less than 0.01 times the hypothetical amount that can be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, are to react.*
- *Any calculated changes in core geometry will be such that the core remains amenable to cooling.*
- *After any calculated successful initial operation of the ECCS, the calculated core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended time required by the long-lived radioactivity remaining in the core.*

These criteria were established to provide significant margin for ECCS performance following a LOCA. The staff found that these acceptance criteria were consistent with the requirements of 10CFR50.46 (b)(1) – (b)(5) for ECCS performance and, therefore, were acceptable.

Westinghouse performed the SBLOCA analyses with the NOTRUMP code for eight cases:

1. *25.4-cm (10-inch) cold-leg break*
2. *double-ended CMT balance line break (17.8-cm [7-inch] in equivalent diameter)*

3. *double-ended rupture of direct vessel injection line (10.2-cm [4-inch] in equivalent diameter)*

4. *5.08-cm (2-inch) cold-leg break in the Passive Residual Heat Removal (PRHR) loop*

5. *5.08-cm (2-inch) cold-leg break in the CMT loop*

6. *6.14-cm (2.4-inch) inadvertent opening of ADS flow paths*

7. *1.27-cm (0.5-inch) cold-leg break*

8. *5.08-cm (2-inch) hot-leg break*

Major assumptions made in the SBLOCA analyses were as follows:

- *As required by Appendix K to 10CFR Part 50, the initial core power is assumed to be 102 percent of the nominal core power and the ANS-1971 decay heat plus 20 percent is used.*

- *Accumulators are initiated at a pressure of 4.83 MPa (700 psia).*

- *The PRHR is opened with the maximum delay of 21.2 seconds after initiation of an "S" signal to delay the cooling capability of the heat exchanger to the RCS.*

- *The "S" signal is actuated when the pressurizer pressure decreased below 11.72 MPa (1700 psia). The CMT isolation valves are opened with the maximum delay of 21.2 seconds after the "S" signal to minimize its contribution to RCS inventory in the initial stage of larger SBLOCAs. The main feedwater isolation valves are ramped closed between 5 and 10 seconds after the "S" signal. The RCPs are tripped 16.2 seconds after the "S" signal.*

- *The ADS actuation signals are taken from the lower of the two CMT levels to be consistent with the CMT actuation delay feature.*

- *The SG isolates (by closure of the turbine stop valves) 1 second after the reactor trip signal to maximize the SG secondary energy. The SG safety valves actuate when the SG pressure reaches 7.58 MPa (1100 psia).*

The results showed the 10-inch break case to be the limiting SBLOCA case with a calculated PCT of 453°C (848°F). The analytical results met the acceptance criteria of 10CFR50.46, Appendix-K with large margins to the acceptance limits. As a result, the staff concluded that the SBLOCA analysis was acceptable.

3.2.1.2 Phenomena Identification and Ranking Table

It was important to identify all physical phenomena that would occur in the AP600 under accident conditions of interest to ensure that the important physical processes and phenomena were modeled. One method of identification was through the development of a PIRT. The PIRT methodology provides a framework where physical processes and phenomena in a specific hardware geometry under anticipated

accident sequences are first identified and then ranked in terms of their importance to the course of the analysis. A PIRT is generally developed from expert opinions provided by a group of knowledgeable analysts. The use of a group of experts, rather than a single analyst, increases the chances that all important phenomena have been identified and included in the PIRT, and that the rankings have accurately characterized each specific phenomena as being of high, medium, or low importance to the integral quantities of interest. A properly established PIRT acts as a road map through a transient, identifying and ranking the important phenomena and functions necessary to predict and deal with each phase of a transient. The PIRT for AP600 SBLOCA can be found in WCAP-14807.

The staff also developed a PIRT as part of the review and confirmatory process. The NRC PIRT for the AP600 SBLOCA is documented in a report from Idaho National Engineering Laboratory, INEL-94/0061, Revision 1. The PIRT prepared by Westinghouse divided the SBLOCA into four intervals: (1) blowdown, (2) natural circulation, (3) ADS blowdown, and (4) In-containment Refueling Water Storage Tank (IRWST) injection cooling. Within each interval, the specific hardware and phenomena were evaluated as having high (H), medium (M), or low (L) importance. The NRC PIRT contained five intervals. The hardware functions and phenomena within two of the NRC PIRT intervals, "Passive Decay Heat Removal" and "CMT Drain to ADS Actuation," were accounted for in the Westinghouse PIRT interval "Natural Circulation." Therefore, all hardware functions and phenomena were accounted for. The Westinghouse PIRT and the NRC PIRT were deemed to be comparable.

Westinghouse also submitted a list of the important phenomena and hardware items identified in the PIRT with a description of the test program, and planned benchmark and assessment calculations which would provide supporting validation for the plant analyses.

The staff compared the Westinghouse and NRC PIRTs and found that all high- and medium-ranked phenomena are captured both in the PIRTs and in the testing program. As a result, the NRC staff found the Westinghouse PIRT to be applicable to the AP600 passive reactor design.

Refer to Section 2.4 of the AP1000 PIRT and Scaling Assessment (Reference 10) for the details of the SBLOCA PIRT.

3.2.1.3 Evaluation of the NOTRUMP Analytical Models

NOTRUMP is a general (variable) nodalization code. Plant models are constructed from generalized control volumes (fluid and metal nodes), flow links, heat sources, and heat sinks. The nonequilibrium thermodynamics and hydraulics include several drift-flux options to calculate relative vapor/liquid velocities (slip). Fission heat is calculated using reactivity and reactor kinetics. The code has an extensive number of forced- and natural-convection heat transfer correlations covering the spectrum of the boiling curve.

Critical flow correlations available include the Moody model, a modified Zaloudek model, and the Murdock-Baumann model. Special-purpose models include flooding, bubble rise, mixture level tracking, a continuous contact flow link, variable flow links, a horizontal stratified flow model, and externals which provide the user flexibility to "program" user specific modifications. Component models include an accumulator, a centrifugal pump, steam separators, and a fuel rod model. The user has available control volumes, flow paths, and heat slabs which can be used to control pressure, enthalpies, mixture levels,

mass flows, and heat fluxes as a function of time. Simple valves are simulated as input flow loss coefficients.

Application of the approved NOTRUMP computer code to the AP600 passive reactor design required a number of modifications, or enhancements, to the basic NOTRUMP models. Nineteen modifications were made to the models as follows:

1. *Add the SIMARC drift-flux model.*
2. *Modify the drift-flux correlations.*
3. *Recast the momentum equations for net volumetric flow.*
4. *Add the NOTRUMP EPRI/Flooding Drift-Flux Model.*
5. *Modify contact coefficients.*
6. *Add internally calculated liquid reflux flow links.*
7. *Add mixture overshoot logic.*
8. *Add implicit treatment of bubble rise.*
9. *Modify the pump model.*
10. *Add implicit treatment of momentum equation gravitational head terms.*
11. *Modify horizontal flow drift-flux levelizing model.*
12. *Add an Unchoking Model.*
13. *Add Shah condensation correlation.*
14. *Add Zuber critical heat flux correlation.*
15. *Change the two-phase friction multiplier.*
16. *Add Henry/Fauske model and homogenous equilibrium model.*
17. *Modify fluid node stacking logic.*
18. *Modify transition boiling correlation solution.*
19. *Revised code numerics.*

All models described above were reviewed by the staff and deemed to be acceptable for analysis of the AP600 SBLOCA.

3.2.1.4 Evaluation of the NOTRUMP AP600 Component Models

In addition to the NOTRUMP model modifications, hardware-specific component models were added to represent AP600-unique hardware features. Component model additions comprise the following:

- *ADS*
- *CMT*
- *PRHR Heat Exchanger (HX)*
- *IRWST*

The following is a brief summary of the component models added to the NOTRUMP code.

Automatic Depressurization System

The ADS is designed to depressurize the RCS to values near the prevailing containment pressure to enable gravity injection from the IRWST. Three stages of the ADS come off the top of the pressurizer; the

fourth-stage ADS paths are connected to the hot legs. The first stage ADS is actuated when 33 percent of a CMT liquid has drained, resulting in the depressurization of the plant via the ADS valves to the IRWST. The second and third stage ADS valves open on the basis of timers that are started with the actuation of the first stage and also discharge to the IRWST. If the CMTs continue to drain, the fourth stage ADS will actuate when 80 percent of the liquid has drained from a CMT. The fourth stage ADS valves, located on the hot legs, open directly to the containment to facilitate depressurization to the containment pressure.

For critical flow, the NOTRUMP code compared well with flow data from the ADS tests, which indicates that the critical flow models in NOTRUMP perform acceptably for the calculation of flow through the ADS valves. This is a highly ranked PIRT item. However, NOTRUMP tends to underpredict the upstream piping pressure drop in the tests and overpredicts the pressure drop of the ADS valve. When the flow in the ADS valve is choked, NOTRUMP overpredicts the pressure drop. However, the overall ADS system pressure drop is predicted well, resulting in correct prediction of ADS choked flow. This situation raised a concern about the models and how they affect the fluid conditions at the entrance to the ADS piping. Westinghouse reviewed the ADS 1-3 test data and determined that the data reduction was performed correctly. The staff reviewed a comparison of the pressure, flow rate, and timing results for the SPES and OSU tests, and the responses to the staff's concerns and found them acceptable.

Core Makeup Tank

There are two CMTs connected to the RCS by normally open isolation valves on the cold-leg balance lines and normally closed isolation valve on the CMT discharge lines. The CMTs provide high-pressure, gravity-driven, borated coolant injection into the RCS to provide reactivity control and core cooling. The CMT discharge valves open on a safety (S) signal and remain open. During normal operation, the CMTs and the cold-leg balance lines are completely filled with liquid.

The AP600 CMTs are new hardware designed subsequent to the guidance of NUREG-0737. WCAP-14807 documents the results of comparisons between the data obtained in the tests and the NOTRUMP calculations. The comparisons indicate that NOTRUMP, when using a multi-node CMT model, gives a reasonable prediction of the temperature distribution within the CMT. Also, the CMT pressure is predicted reasonably well and the outflow of the CMT is predicted within the error bounds on the data. Because the NOTRUMP code does not have a thermal stratification model, the predicted temperature of the injected CMT fluid is usually higher than the measured temperature, and the start of CMT draining is frequently delayed. Each of these inaccurate code predictions was deemed to be conservative; therefore, the staff found the NOTRUMP model for the CMTs acceptable for evaluations of the AP600 SBLOCA.

Passive Residual Heat Removal Heat Exchanger

The PRHR system is a C-shaped, single-pass, downflow heat exchanger, submerged in the IRWST. The system inlet connects to the top of the horizontal hot-leg section containing the pressurizer loop. The system outlet connects to the bottom of the pressurizer loop steam generator outlet plenum. Normally closed isolation valves open to actuate the system on a safety (S) signal.

The PRHR HX is immersed in the IRWST. Heat transfer is modeled using the standard NOTRUMP heat transfer correlations plus, on the inside of the tubes, the Shah correlation, as discussed above for condensation modeling, and the Lienhardt and Dhir modified Zuber correlation for critical heat flux on the IRWST side of the tubes.

After review of the integral system assessments included in WCAP-14807, the staff notes that the PRHR HX heat transfer calculated by NOTRUMP for the SPES and OSU transients is lower than that measured in the experiments. This is a conservative result, and therefore, the staff accepted the NOTRUMP PRHR model for analysis of the AP600 SBLOCA.

In-Containment Refueling Water Storage Tank

The IRWST provides a source of water for gravity feed injection into the RCS once RCS pressure has been reduced to values near the containment pressure. The IRWST also serves as a heat sink for the removal of heat via the PRHR discussed above and a discharge reservoir for the first three stages of the ADS. Condensation of steam in the containment provides a long-term source of water to the IRWST, which can then return to the RCS. Although not part of the IRWST, the containment sump provided a second source of gravity fed coolant injection into the RCS over the long term.

The NOTRUMP analyses were performed for a range of SBLOCAs for both the SPES and OSU integral facilities. The comparisons between NOTRUMP calculations and experimental data for the SPES and OSU tests, documented in WCAP-14807, show acceptable agreement. The IRWST injection line flows, outlet flows, and PRHR inlet and outlet temperatures were predicted reasonably well. The staff found the NOTRUMP IRWST model acceptable for analysis of the AP600 SBLOCA.

3.2.1.5 Code Qualification

Qualification, or assessment, of the NOTRUMP code and its models was carried out in three areas: (1) benchmark calculations, (2) separate-effects tests, and (3) integral systems tests. The combination of benchmark calculations, separate-effects tests, and integral systems tests, when properly applied, leads to overall conclusions regarding the ability of a computer code to adequately predict the behavior of a nuclear power plant subjected to upset and accident conditions. Because no single test captures all of the relevant phenomena, it is necessary to utilize all three categories to adequately cover the phenomena of interest. The three categories are discussed below.

Benchmark Calculations

Benchmark calculations are useful to demonstrate that logic interactions do not result in numeric instabilities, or physically unrealistic results. In general, these benchmark problems consist of thought problems (hypothetical problems not on the basis of data from an actual test facility) and simple nodal models to verify a particular code function or single phenomenological behavior. Assessment of the NOTRUMP code via benchmark calculations was performed for areas involving changes to the previously approved code such as the reactor coolant pump models, plus those areas involving logic changes and additions to the code. Extensive logic modifications were made, as previously discussed, involving mixture level overshoot, fluid node stacking, and bubble rise. Many of these logic models interact during the calculation of the SBLOCA. Separate from the assessment of the overall performance

of these models in predicting the integral test facility behavior, benchmark calculations were performed for the following models:

- *NOTRUMP EPRI/flooding drift-flux model*
- *Horizontal drift-flux levelizing model*
- *Net volumetric flow-based momentum equation*
- *Implicit treatment of gravitational head*
- *Implicit treatment of bubble rise*
- *Pump model*
- *Fluid node stacking logic*

In WCAP-14807, Westinghouse used the benchmark calculations to demonstrate that NOTRUMP calculated results for each case that agreed with the logical expectations for that case. No unrealistic model interactions were uncovered and no numeric instabilities were encountered. The staff reviewed each of the benchmark calculations performed and found the models acceptable for the analysis of AP600 SBLOCA events.

Separate Effects Tests – Two-Phase Level Swell

Assessment of the NOTRUMP code against separate-effects tests permits the isolation of individual models within the code such that the capabilities of the model can be determined while remaining within the context of the code.

The two-phase level swell can be an important phenomenon during an SBLOCA. Although the AP600 SBLOCA is not predicted to result in core uncovery, a two-phase mixture will exist in the upper vessel regions and therefore, the code must be capable of predicting the location of a two-phase level. The two-phase level swell model extensions to accommodate the low pressures anticipated in the AP600 SBLOCA were assessed by comparisons with data obtained in three test facilities. Westinghouse analyzed tests from the G-2 test program, from the General Electric (GE) level swell test program, and from the Achilles systems test program.

The three test programs, given above, were chosen to encompass the anticipated pressure and flow conditions in the AP600 design for the assessment of the two-phase level swell model in NOTRUMP. The GE tests covered the intermediate pressure of 6,894.7 kPa (1,000 psia); the G-2 tests covered the range from 5,515.8 kPa (800 psia) to 101.3 kPa (14.7 psia); and the Achilles tests provided integral system data at 101.3 kPa (14.7 psia) and 202.6 kPa (29.4 psia) pressure. In addition to the coverage of the anticipated pressure range, the three test programs provided data for different scale facilities.

The assessment of two-phase level swell under the anticipated pressure and power conditions of the AP600 SBLOCA was difficult because of the lack of low-pressure two-phase level swell data. All data currently available are from test facilities with flaws in the tests and data collection that make it necessary to make assumptions in the computer code modeling of the facilities and tests. Thus, there are no known ideal test results for assessment of the two-phase level swell capabilities of the code at low pressure. When reasonable assumptions are made to account for the facility problems noted in this section, the NOTRUMP code does an acceptable job of predicting two-phase level swell. The results of the assessments, presented in WCAP-14807, indicate that NOTRUMP underpredicts the mixture level

over a wide range of thermal-hydraulic conditions that may be found during AP600 SBLOCAs. The predicted level is consistently conservative or within the test data uncertainty. The staff finds that the tests used in the assessment of NOTRUMP two-phase level swell sufficiently test the code's capability to permit the judgment that the NOTRUMP code adequately predicts two-phase level swell at low system pressure.

Integral Systems Tests

Integral systems tests permit an assessment of the entire code, including all pertinent models, acting as a unit to predict the full system behavior. Westinghouse analyzed selected SPES and OSU integral tests for the final verification and validation effort. Comparison between several of the related tests assists in understanding the effects of scale on the analysis results.

Assessments were performed to compare the results of NOTRUMP calculations to data from SPES and OSU tests for a variety of transients covering a wide range of break sizes and locations. The NOTRUMP code was found to provide reasonable predictions of the highly ranked PIRT phenomena, including the following:

- *Pressurizer pressure and level*
- *Core inlet and outlet temperatures*
- *CMT injection flow rates and collapsed liquid level*
- *The steam generator collapsed liquid level as well as pressure and temperature*
- *The cold leg balance line levels*
- *The upper plenum and upper head collapsed liquid levels*
- *The PRHR inlet and outlet temperatures*
- *The break flow rate*

An exception to the above-noted acceptable results is the double-ended guillotine break of a DVI line. The calculated results for core level during a double-ended guillotine break of a DVI line were nonconservative (higher) than the measured value. The Westinghouse explanation of the differences in core and downcomer behavior in the DVI line break is primarily because of the one-dimensional nature of NOTRUMP. The test data indicate that a two-dimensional temperature pattern develops in the downcomer that NOTRUMP is not able to predict. This allows portions of the downcomer to remain saturated and to flash when ADS 1-3 open. Less mass is then stored in the downcomer. Also, the vapor generated in the core exits through the broken DVI line and not through the intact DVI line. NOTRUMP predicts vapor exiting by both paths. The staff believes that the AP600 will perform in a manner that is more similar to the behavior of the test facilities than to the behavior predicted by NOTRUMP. This discrepancy is resolved by the time ADS 1-3 blowdown is completed, as evidenced by good agreement between the measured and predicted core levels in both the SPES and OSU tests. Thus, the discrepancy does not adversely affect the prediction of core level. Although the NOTRUMP code is unable to predict the two-dimensional behavior of the design, the staff concludes that this discrepancy is acceptable because the core does not uncover in either the tests or the calculations. Core heatup does not occur in either case. A two-dimensional analytical capability would be desirable, but would not appreciably change the results.

3.2.1.6 Regulatory Compliance

Following the accident at TMI, the NRC focused attention on the SBLOCA and proposed revisions to the methods and analyses performed to better demonstrate compliance with the requirements set forth in 10CFR50.46. With regard to Westinghouse-designed PWRs, the NRC outlined technical issues in NUREG-0611 ("Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse Designed Operating Plant") regarding the capabilities of the WFLASH computer program used to simulate the reactor coolant response to a SBLOCA. WFLASH was an early methodology that Westinghouse developed for simulation of SBLOCA response. In NUREG-0611, the staff identified specific models in the WFLASH computer code that were considered deficient. Furthermore, the NRC issued NUREG-0737, Section II.K.3.30 to clarify the post-TMI requirements regarding SBLOCA modeling. In essence, Section II.K.3.30 of NUREG-0737 recommends that licensees of Westinghouse-design PWRs revise their SBLOCA models in accordance with the guidelines specified in NUREG-0611, or justify continued acceptance of the current model. Section II.K.3.31 further recommends that each licensee submit a new SBLOCA analysis using an approved evaluation model that meets the criteria of NUREG-0737, Section II.K.3.30.

In response to these requirements, Westinghouse developed the NOTRUMP code for reference in the new SBLOCA ECCS evaluation model calculations. As such, the NOTRUMP code was developed to overcome the deficiencies identified in the WFLASH computer program while also addressing the post-TMI requirements. Following NRC review, the NOTRUMP code was approved for evaluating SBLOCA response in Westinghouse-design PWRs.

10CFR Part 50, Appendix K

Westinghouse modified the approved NOTRUMP code for application to the AP600 design for analysis of the SBLOCA in compliance with the requirements of 10 Part CFR 50, Appendix K. Of the many requirements specified in Appendix K, only one refers both to portions of NOTRUMP that have been modified and to phenomena that are anticipated in AP600 SBLOCAs. 10CFR Part 50, Appendix K, Section C.2, requires that the frictional loss in pipes and other components, including the reactor core, be calculated using models that include realistic variation of friction factor with Reynolds number, and realistic two-phase friction multipliers that have been adequately verified by comparison with experimental data, or models that prove at least equally conservative with respect to maximum clad temperature calculated during the hypothetical accident. Appendix K then specifies acceptable correlations.

The friction factor calculations within the AP600 version of NOTRUMP are largely the same as found in the approved version of NOTRUMP with the exception of some smoothing and an extension of two-phase multipliers down to atmospheric pressure. The effectiveness of this modified model was evaluated and documented in WCAP-14807. The outcome of these evaluations demonstrated that the NOTRUMP code results agreed reasonably well with data from two-phase level swell experiments (G-2, GE, and Achilles). The code results were also reasonable for the SPES and OSU test assessments. The staff found the frictional loss model in NOTRUMP acceptable for the analysis of AP600 SBLOCA.

NUREG-0611 and NUREG-0737

As the motivation to develop the NOTRUMP code arose from the guidance in NUREG-0611 and NUREG-0737, the modifications and applicability of the modified code to AP600 SBLOCA evaluations have been reviewed regarding the following TMI small break modeling concerns:

1. *Provide calculated validation of the SBLOCA model to adequately calculate the core heat transfer and two-phase coolant level during core uncovery conditions.*

   The NRC requested that the heat-up methodologies be compared to the core cooling tests performed by the Oak Ridge National Laboratory (ORNL) at its Thermal Hydraulic Test Facility (THTF). The ORNL tests provide a good database to assess the heat transfer capabilities of a fuel rod subjected to uncovery and the resultant steam cooling conditions that can occur in the upper portion of the bundle. The tests cover a wide range of pressures and rod powers for both transient film boiling and bundle uncovery steam-cooling conditions. Predictions of two-phase level swell as well as the steam cooling convection heat transfer are essential to successful predictions of SBLOCA response.

   Evaluation

   The AP600 integral data did not indicate core uncovery and, therefore, the heatup model was not exercised. The assessments in WCAP-14807 of the G-2, GE, and Achilles tests provided verification of the steam cooling model and level swell in NOTRUMP during core uncovery conditions at the low pressure and low flow anticipated in the AP600 design. The staff found the NOTRUMP code acceptable to evaluate core heat transfer and two-phase level swell in the AP600 SBLOCA.

2. *Validate the adequacy of modeling the primary side of the steam generators as a homogeneous mixture.*

   It is necessary to demonstrate that there is sufficient spatial detail to model the primary and secondary systems to properly account for forward and reverse heat transfer as liquid drains from the primary active tubes. Because the steam generators can act as a heat source following some SBLOCAs, proper accounting for the steam-water behavior and associated depressurization rates is required.

   Proper accounting of the annular and slug flow regimes as they may occur in the steam generators should be incorporated into the modeling of this region. One must, therefore, ensure that the flow region behavior in the generators is consistent with the heat transfer conditions throughout the transient. If there is a potential for flooding or "hold-up" of the liquid in the generators, then the hydraulic model should also account for this behavior. Also, if a stratified flow model is used in the hot-leg piping, this flow regime should be justified.

Evaluation

Steam generator heat transfer has a secondary role in AP600 since the PRHR system functions along with the ADS to control system pressure and depressurize the plant. This low importance of steam generator heat transfer is reflected in the low PIRT ranking. Nevertheless, the models contained in NOTRUMP are applicable to the AP600 plant performance.

In the AP600, the PRHR HX functions in a manner similar to the steam generators as a major heat removal system. The PRHR uses the same models; thus, the staff considered its modeling under this item. The PRHR heat transfer was not properly predicted in the SPES and OSU comparisons since the code was unable to predict the correct outlet temperature. The NOTRUMP code tends to overpredict the outlet temperature. This result is conservative; therefore, the staff found the NOTRUMP model for the steam generator and PRHR system acceptable.

NOTRUMP contains provisions for stratified and dispersed flow regimes in the loop piping and steam generators. The flooding models in the code capture the potential for liquid hold-up in the loop and steam generators should steam velocities be sufficient to entrain and limit drainage in the loops. The PIRT ranking for flow-regime-related phenomena is low, so that this phenomenon does not appear to have a significant impact on AP600 performance. Comparisons between the NOTRUMP calculated fluid conditions and the measured fluid conditions from SPES and OSU tests demonstrated that the models used in NOTRUMP for the stratified and dispersed flow regimes are appropriate. The staff found the NOTRUMP models for the stratified and dispersed flow regimes in the hot leg piping acceptable for analysis of the AP600 SBLOCA.

3. *Validate the condensation heat transfer model and effects of noncondensable gases.*

The condensation correlation used in the blowdown hydraulics code must be justified as to the applicability to the two-phase flow conditions in the active tubes of the steam generators. The need for a best-estimate correlation was stressed as opposed to empirical relationships containing "conservatisms." Particular emphasis should be placed on the applicability of the correlations to U-tube steam generators since most correlations used to date are on the basis of flat-plate geometries. Noncondensable gases should also be accounted for.

Evaluation

Westinghouse added the Shah condensation correlation to the NOTRUMP condensation correlation package previously approved. The staff found the addition of the Shah correlation acceptable for analysis of the AP600 SBLOCA.

The NOTRUMP code can not calculate the effects of the noncondensable gases injected into the primary coolant system during the AP600 SBLOCA. The presence of noncondensable gases is of concern because of the possible degradation in performance of the PRHR HX for system depressurization and heat removal. During the conduct of the test program, noncondensable gases entered the system but were not tracked as they moved through the system. The gases either exited the system or were found to end up in the PRHR HX or the CMTs. It was noted that the noncondensable gases entered the PRHR HX late in the transient, when the PRHR system no

longer had a significant role in heat removal. At this point in the transient, the noncondensable gases do not appear to have a detrimental effect on the system. Since the noncondensable gases do not play a role in the AP600 SBLOCA, the staff accepted the NOTRUMP code for evaluation of AP600 SBLOCA in spite of its inability to calculate the effects of noncondensable gases. This position will be re-evaluated if scenarios are found which cause noncondensable gases to reach the PRHR HX while it is actively removing heat from the primary system.

4. *Demonstrate, through noding studies, as part of the sensitivity studies, the adequacy of the SBLOCA model to calculate flashing during system depressurization.*

   Evaluation

   The adequacy of the NOTRUMP code to model the system effects, as well as local fluid conditions, during the AP600 SBLOCA is demonstrated through the consistency in noding between the scaled, integral system test facilities (SPES-2 and OSU) and the AP600 design. The staff found the noding acceptable for analysis of the AP600 SBLOCA on the basis of analyses performed on the noding differences used in the PRHR and downcomer models for the test facilities and AP600.

5. *Validate the polytropic expansion coefficient applied in the accumulator model.*

   Evaluation

   The accumulator model was not changed from the approved NOTRUMP code to the AP600 version of the code. Also, the accumulator in the AP600 design is similar to that employed in the current generation operating Westinghouse PWRs. The staff therefore accepted the model as it was applied to the AP600 SBLOCA.

6. *Break discharge model*

   Since Appendix K to 10CFR Part 50 requires use of the Moody critical flow model, the blowdown hydraulic code must contain this methodology.

   Evaluation

   The Moody critical model is used in the NOTRUMP code as specified by Appendix K (10FR Part 50) for saturated break flow. It is noted that the flow rate out the ADS is being predicted through the addition of the Henry/Fauske and HEM critical flow models previously discussed. Because the ADS is a depressurization system and not an actual break, this is considered acceptable based in part on the precedents established by the depressurization system used in the boiling water reactors. The staff found that Westinghouse demonstrated that the treatment of the ADS is conservative.

7. *Validate the SBLOCA model with loss-of-fluid test (LOFT) facility tests L3-1 and L3-7. In addition, validate the model with the Semiscale S-UT-08 experimental data.*

   There is a need for integral as well as separate-effects test comparisons. The NRC identified the LOFT and Semiscale integral-system tests which should be included as part of the code verification process. These tests include LOFT and Semiscale integral system tests addressing SBLOCA transients, including an examination of the continued operation of the main coolant pumps on the system response following initiation of a small break (L3-6). Semiscale S-07-10D was also identified as an integral test that could be used in the benchmarking of codes against a SBLOCA transient where long-term core uncovery was simulated.

   Evaluation

   The purpose of this requirement was to demonstrate the ability of the code to adequately deal with plugging and clearing of the steam generator to the reactor coolant pump loop seal. The AP600 design eliminated the loop seal, as well as placing the steam generators entirely above the hot-leg reactor vessel nozzle, which is above the reactor core. Accordingly, the staff found that this requirement was not applicable to the analysis of the AP600 SBLOCA since there is no loop seal to prevent the steam generator tube contents from flowing into the reactor vessel. The staff noted that Westinghouse provided other integral effects assessments of NOTRUMP that address the highly ranked PIRT items.

   The staff was aware that Westinghouse submitted modifications to the NOTRUMP code incorporating a condensation model on the basis of results of the COSI safety injection (SI)/steam condensation experiments. The COSI test facility is a scaled representation of the cold-leg and SI injection ports in a Westinghouse designed PWR. The pressure range covered by the COSI tests is outside of the range of interest for the low-pressure conditions expected in the AP600 SBLOCA. In addition, the AP600 design uses direct vessel injection for SI. Accordingly, the staff position was that the COSI condensation model is neither applicable nor acceptable for evaluation of the AP600 SBLOCA.

In addition to the above modeling concerns, NUREG-0737 recommendations indicate that the effect of the operation of the main coolant pumps on SBLOCA response should be assessed. The AP600 design is such that a safety-grade, single-failure- proof, reactor coolant pump trip is provided. As such, Westinghouse was not required to evaluate AP600 performance with the main coolant pumps operating.

3.2.1.7 ACRS Review

Two meetings were held with the ACRS Thermal-Hydraulic Subcommittee for review of the NOTRUMP code. Those meetings resulted in numerous additional review items and concerns. As a result the staff required that Westinghouse fully document the code numerics, providing detailed derivations of all equations modified or changed from the source form to the difference form as applied in the code. This was in addition to fulfilling the commitments Westinghouse made during the meetings. Subsequently, Westinghouse documented responses to the following six issues:

1. *Momentum Flux – Deficiencies were benchmarked against additional detailed calculations using actual two-phase flow equations that include the effects of compressibility and the condition of constant entropy.*

2. *ADS 1-3 – The test data analysis report was revised to show that the data reduction was performed correctly.*

3. *Entrainment – Entrainment was considered as part of the overall scaling and IRWST-level penalty development.*

4. *IRWST-Level Penalty – A multiloop scaling analysis was performed for the time period of ADS-4 and IRWST draining. The basis for ADS flow was justified, along with ADS-4 flow affected by entrainment of liquid and the corresponding effect on the pressure loss as a result of two-phase flow.*

5. *Pressurizer Surge Line Flooding – An evaluation similar to that applied to the IRWST level penalty was performed.*

6. *Noding – Additional justification was provided for the basis used which differs from the accepted approach developed under the CSAU work. This applies in particular to the PRHR and downcomer.*

Several of these items involved phenomena that are not well represented or modeled in NOTRUMP, because of the structure of the code. Nonetheless, overall code calculations of the plant's performance showed large margins to licensing limits and all issues were addressed in a conservative fashion. Therefore, the staff concluded that these issues did not alter the staff's determination that NOTRUMP was suitable for analyzing the behavior of the AP600.

3.2.1.8 Conclusions

The NOTRUMP computer code was developed by Westinghouse to assess the consequences of an SBLOCA. The code was modified through introduction of model additions and changes in 18 of the approved code's models. In addition, component models for the ADS, CMT, PRHR HX, and IRWST were added to make the code applicable to the AP600 passive reactor design. The staff reviewed the code's application to the AP600 SBLOCA, the component test program, and the integral systems tests, which resulted in a large number of RAIs. Westinghouse responded to the RAIs and documented the responses in the NOTRUMP Final Verification and Validation Report, WCAP-14807.

Additional assessment calculations were considered important to the assessment of the level swell models in NOTRUMP. The additional requests for benchmarking were on the basis of the lack of level swell benchmarks provided by Westinghouse in the documentation and the nonconservative predictions displayed by NOTRUMP in several of the SPES and OSU tests. The NOTRUMP code, in these cases, overpredicted the liquid inventory in the core and upper plenum regions of the reactor vessel. Because there was a basic lack of low-pressure data to qualify codes for level-swell phenomena, the staff concluded that additional tests needed to be analyzed for model qualification.

The staff expressed concerns regarding the assessment of many of the models modified in the approved NOTRUMP code. In particular, the changes to the drift-flux models, bubble rise model, and momentum equations significantly alter the two-phase level swell capabilities of the code. An adequate assessment of the two-phase level swell was essential to properly understand the predictions of the code in an SBLOCA situation since it is a depressurizing, two-phase condition. Westinghouse performed numerous assessments of the logic models and the two-phase level swell models to demonstrate the adequacy of the models in predicting two-phase level and void fraction distribution in the AP600 SBLOCA.

In addition, the staff expressed concern about the extensive logic models added to the code to control mixture level, region birthing, etc. It was requested that Westinghouse demonstrate that the interaction of the logic models did not lead to unrealistic results. Also, the staff required Westinghouse to demonstrate that mass and energy were conserved as mass and energy and are redistributed when mixture regions pass through flow links. The "mechanical" movement of mass and energy in these logic schemes suggested that the models be exercised through the benchmark calculations to assure that the conservation laws are not being violated.

Westinghouse added options to NOTRUMP to permit use of the momentum equation in volumetric form and flow partitioning in the analysis of the AP600 SBLOCA. The staff does not consider the "options" added to improve the performance of NOTRUMP in analyzing the AP600 SBLOCA to be options. The staff position is that the "options" added to NOTRUMP for AP600 SBLOCA analyses are required to be used for those analyses.

Because transition boiling was not expected to occur in the AP600 core under SBLOCA conditions, the changes in the numerical solution techniques used in the NOTRUMP heat links when transition boiling is predicted to occur were not reviewed. It was noted by Westinghouse that the core model methodology was unaffected by the change in the transition boiling heat link methodology as these two models are completely separate in the code. Therefore, this revised methodology would not be invoked in the core region of AP600 calculations. Should this revised methodology be applied to core calculations, the review of the modified transition boiling correlation solution scheme would need to be revisited.

The staff noted that the NOTRUMP code could not calculate the effects of noncondensable gases injected into the primary coolant system during the AP600 SBLOCA. Noncondensable gases enter the PRHR late in the transient, when the PRHR HX no longer has a significant role in heat removal. Thus, the noncondensable gases did not appear to have a significant effect on the course of the event. The staff accepted the NOTRUMP code for evaluation of the AP600 SBLOCA in spite of this shortcoming. However, if scenarios are found which cause noncondensable gases to reach the PRHR HX while it is actively removing heat from the primary system, NOTRUMP could not be used to analyze those scenarios.

Notwithstanding the limitations that the staff identified in its review of the application of the NOTRUMP code to analyses of the AP600 design and the conditions that Westinghouse must observe as it applies the code, the staff has confidence that the use of NOTRUMP is acceptable for AP600. This is because the phenomena expected during a SBLOCA are modeled reasonably well in the test facilities, code comparisons with the experiments are reasonable, and they indicate that there are large margins to licensing limits which are unlikely to be challenged by uncertainties in the code models.

In a letter dated February 27, 1998, Westinghouse submitted Revision 4 to WCAP-14807, NOTRUMP AP600 Final Verification and Validation Report. Therefore, with the limitations and conditions described in this report, the staff concluded that the NOTRUMP code had been appropriately modified to include the features necessary to model the AP600 plant and the phenomena expected during an AP600 SBLOCA. Therefore, it could be applied to the AP600 passive reactor design.

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 under-prediction 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 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 successfully be applied to the AP1000 plant design. 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 methods used 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).

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 has been developed for the AP1000 ADS-4 specific flow geometry. The results of this detailed model were then utilized 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. Use of this method more accurately reflect the ADS flow distributions and ultimately the onset of IRWST injection flow. For the AP1000 plant DCD analyses, the detailed momentum flux model

developed for the AP1000 plant design resulted in the required resistance increase determined to be [ ]a,c.

For the scoping study results presented in WCAP-15612 (Reference 9), the ADS-4 flow path resistances were increased at the transition to sub-sonic flow. The loss coefficients utilized were based on the detailed stand-alone momentum flux model results of the ADS-4 flow paths generated for the AP600 plant design. The application of the resistance increase methodology was previously demonstrated to significantly improve the match between the model prediction and the test data. The results of this revised methodology were presented to the ACRS during the May 11th and 12th 1998 Thermal Hydraulic sub-committee meetings. The information presented included comparisons with the OSU 2-inch cold leg break test data as well as AP600 2-inch cold leg break simulations. Subsequent to this meeting, the complete AP600 break spectrum was re-performed utilizing the ADS-4 resistance increase methodology. The results obtained indicated that the ADS-4 resistance increase results were comparable to the IRWST level penalty results, which serve as the basis for the AP600 DSER. While the loss coefficients utilized in the scoping studies (Reference 9) were not AP1000 plant specific, they provide a determination of the overall plant response.

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 mis-prediction 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 under-prediction 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.

3.6 CONCLUSIONS/RECOMMENDATIONS

To appropriately apply the NOTRUMP code to the AP1000 plant design, the deficiencies noted during its application to the AP600 plant are addressed as discussed in Section 3.4.5 and summarized in Table 3-1. The methods described in the previous section address the identified issues in an effective manner thereby allowing Westinghouse to demonstrate the conservative nature of the NOTRUMP code. As such, Westinghouse believes that the NOTRUMP code used for the AP600 test and analysis program is appropriately used in support of AP1000 Design Basis Accident (DBA) analyses. The large plant margins to safety limits observed in the SBLOCA analysis are demonstrated by the AP1000 DCD analyses.

3.7 REFERENCES

1. WCAP-10079-P-A, "NOTRUMP A Nodal Transient Small Break And General Network Code," Meyer, P. E., 1985.

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.

7. 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 (a)(3)(ii)," Letter from H. A. Sepp (Westinghouse) to J. S. Wermiel (NRC), May 12, 2000.

8. NSBU-NRC-00-5972, "NRC Report for NOTRUMP Version 38.0 Changes," Letter from H. A. Sepp (Westinghouse) to S. J. Collins (NRC), June 30, 2000.

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

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

11. WCAP-14292, Revision 1, "Low-Pressure Integral Systems Test at Oregon State University, Test Analysis Report," September 1995, T. S. Andreychek, et. al.

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.

13. 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.

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, under-predicted pressure drop.	Apply IRWST level penalty.[1] Upper bound loss coefficients.	Flow out ADS-4 is over-predicted, 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.	Under-prediction 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 over-predicted 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.	Under-predicted 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.

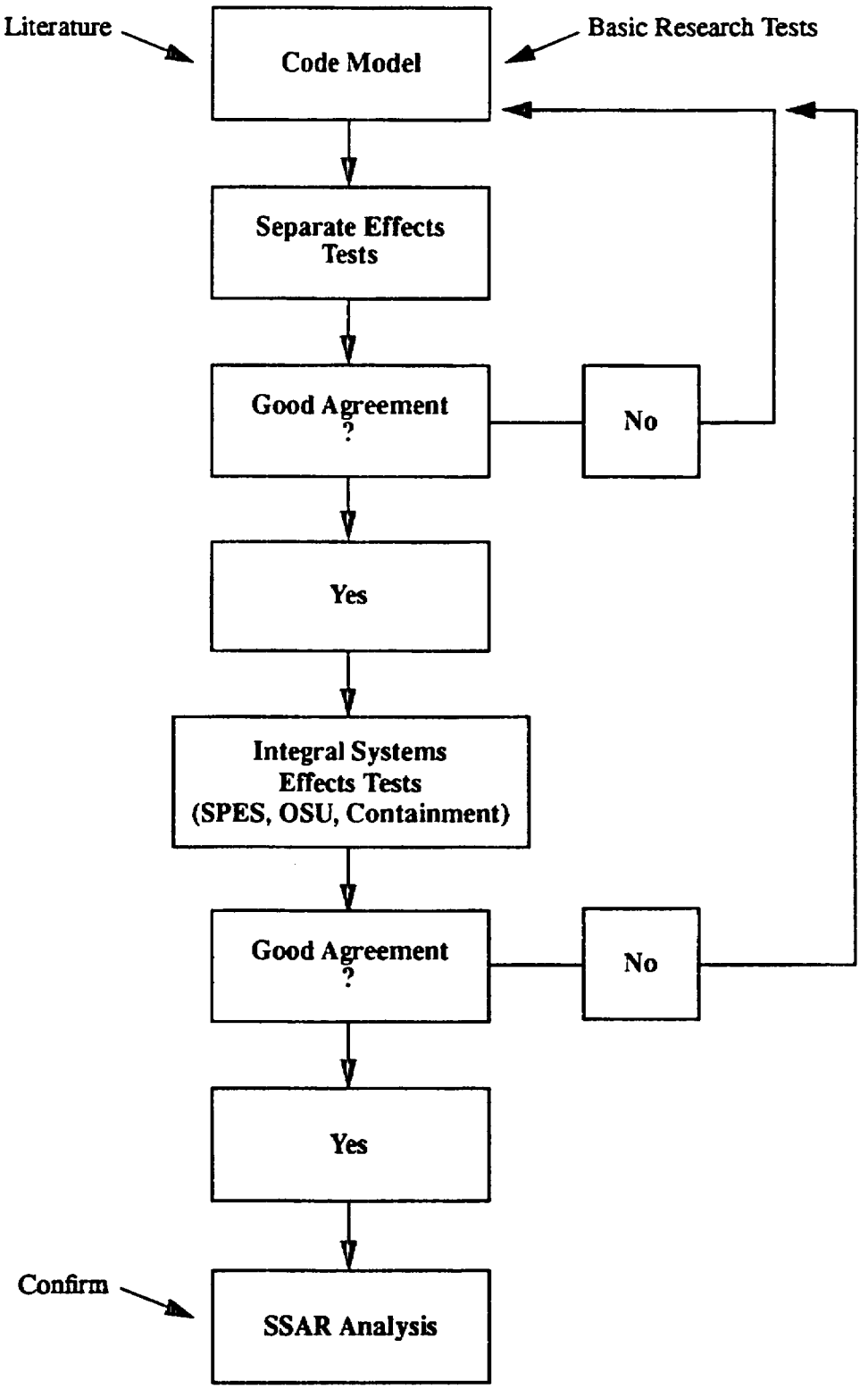


Figure 3-1 Model Development and Verification Process for Code Validation

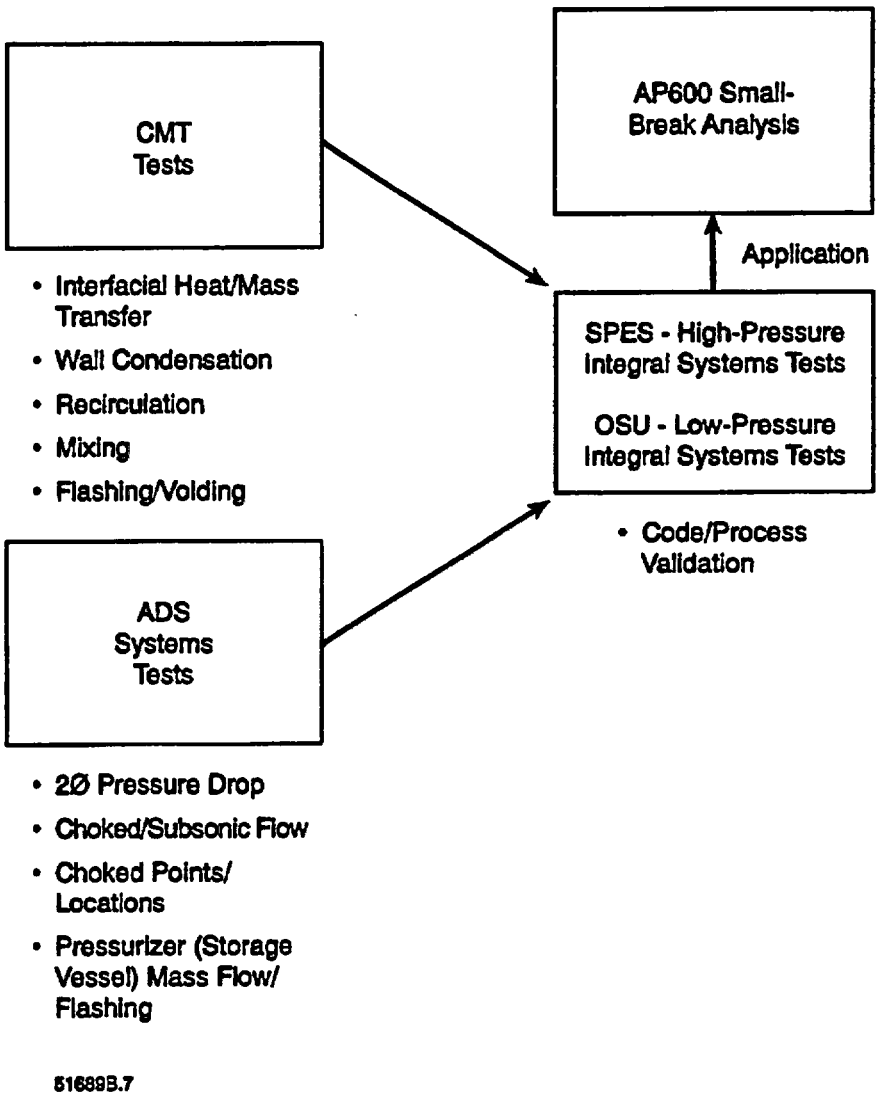


Figure 3-2 NOTRUMP Verification with Separate Effects Tests and Validation with Integral Systems Tests

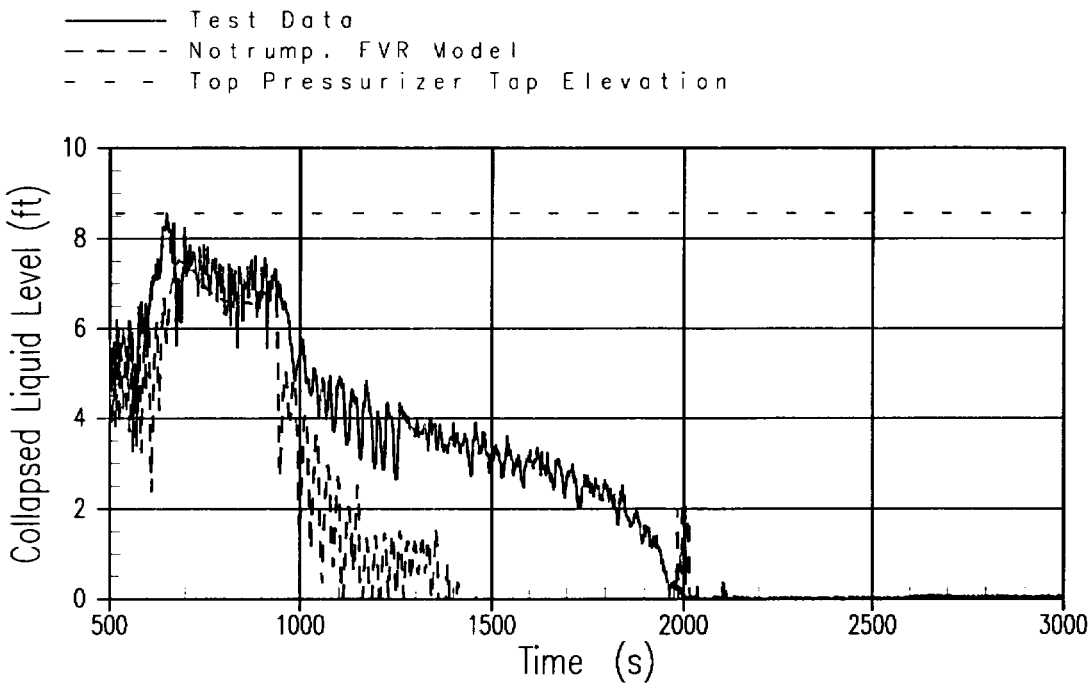


Figure 3-3 OSU Test SB18 2-Inch Cold Leg Break Pressurizer Level (Relative to Bottom Tap)

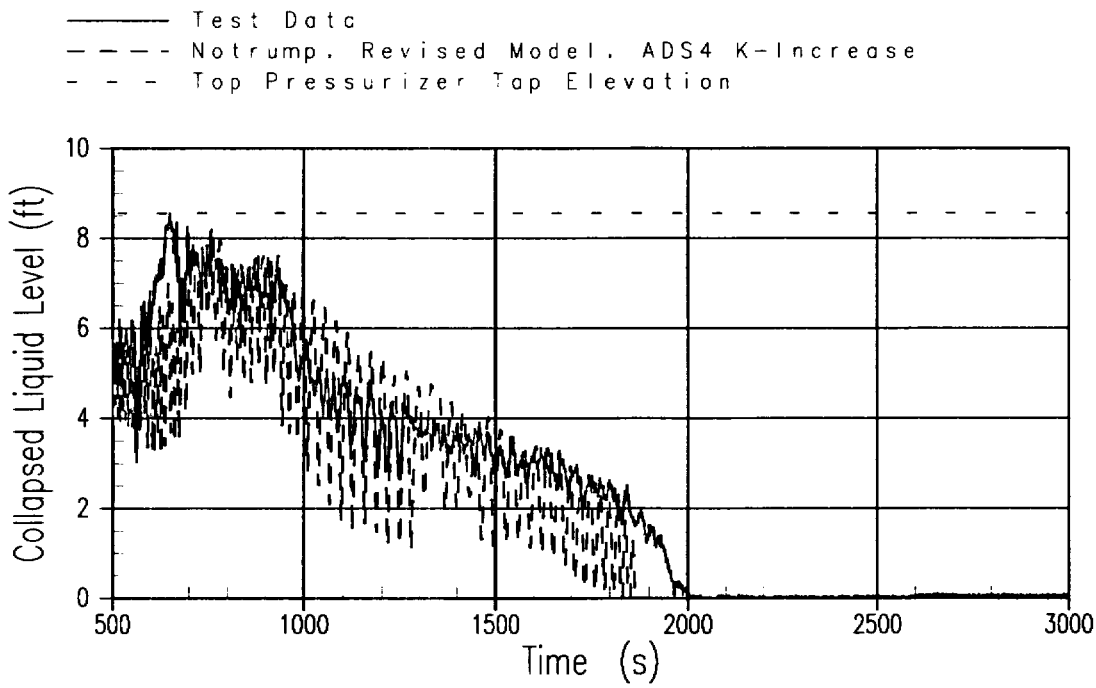


Figure 3-4 OSU Test SB18 2-Inch Cold Leg Break Pressurizer Level (Relative to Bottom Tap)

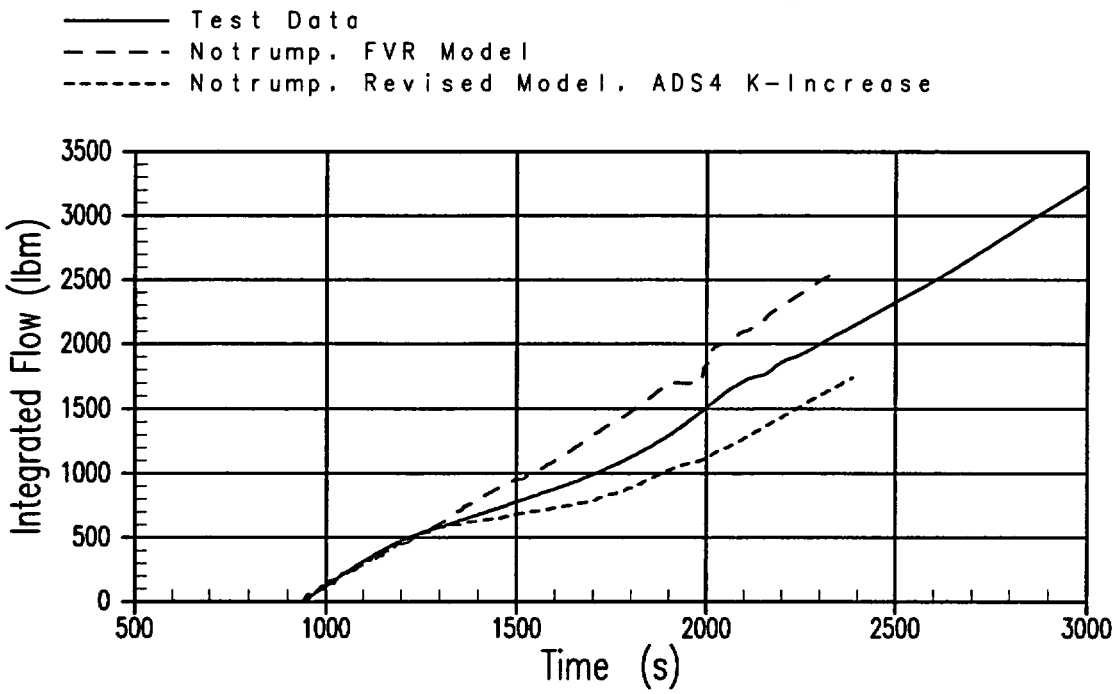


Figure 3-5 OSU Test SB18 2-Inch Cold Leg Break ADS Stage 4 Integrated Flows

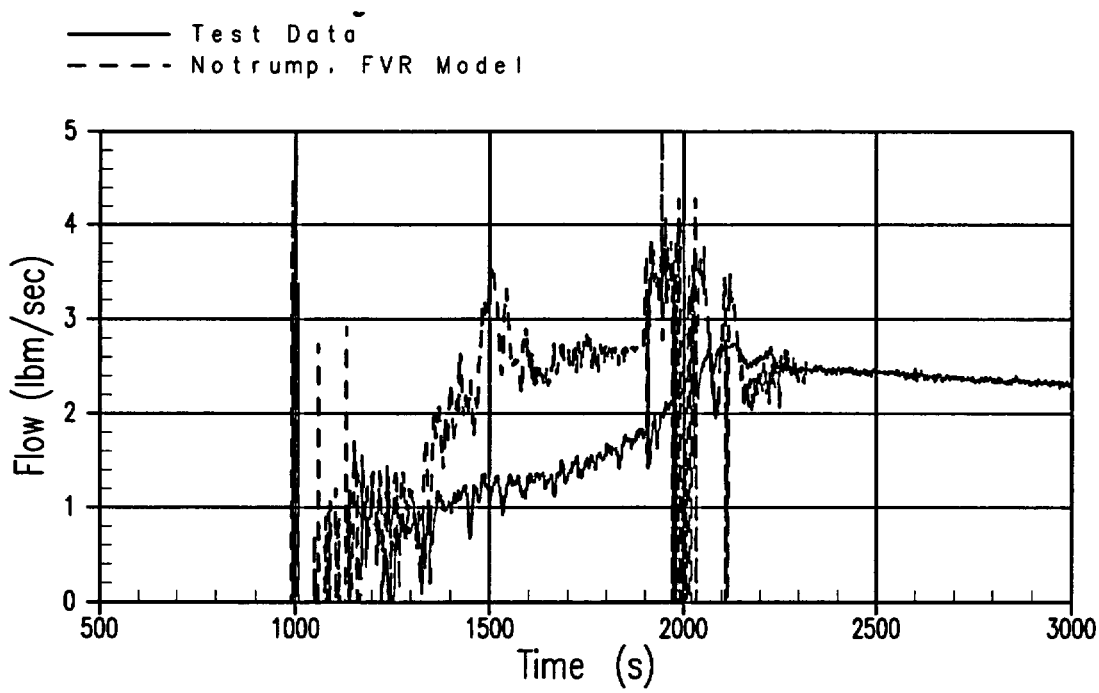


Figure 3-6 OSU Test SB18 2-Inch Cold Leg Break IRWST Injection Line Mass Flow

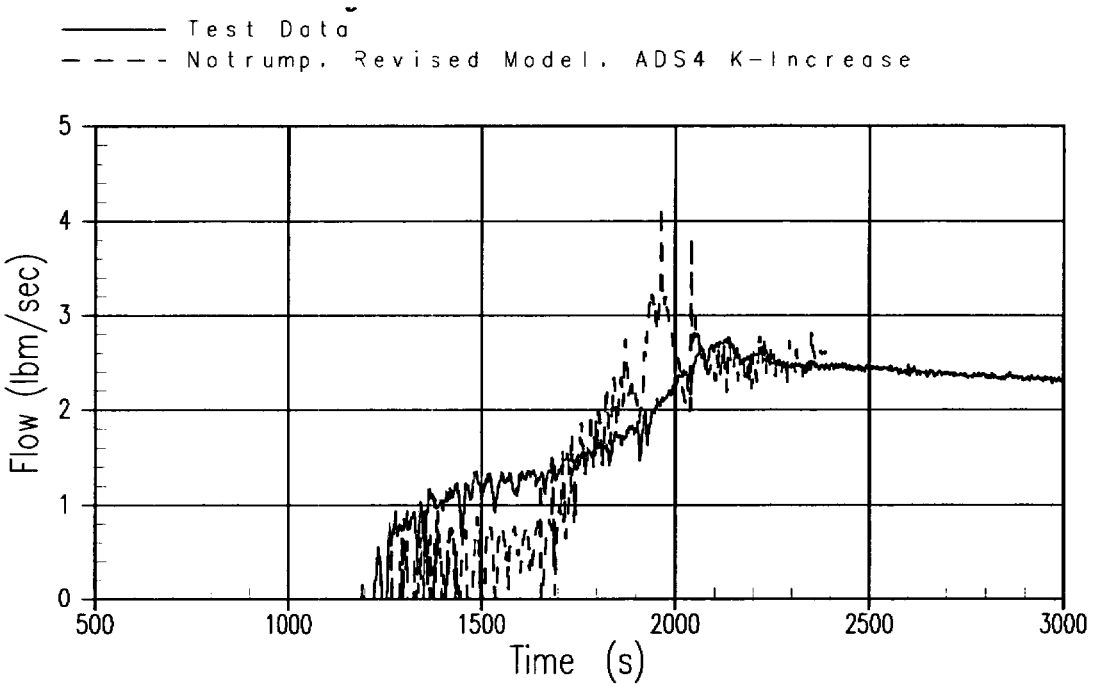


Figure 3-7 OSU Test SB18 2-Inch Cold Leg Break IRWST Injection Line Mass Flow

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 PIRT's 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 apply 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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16. WCAP-15613, "AP1000 PIRT and Scaling Assessment," February 2001.

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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

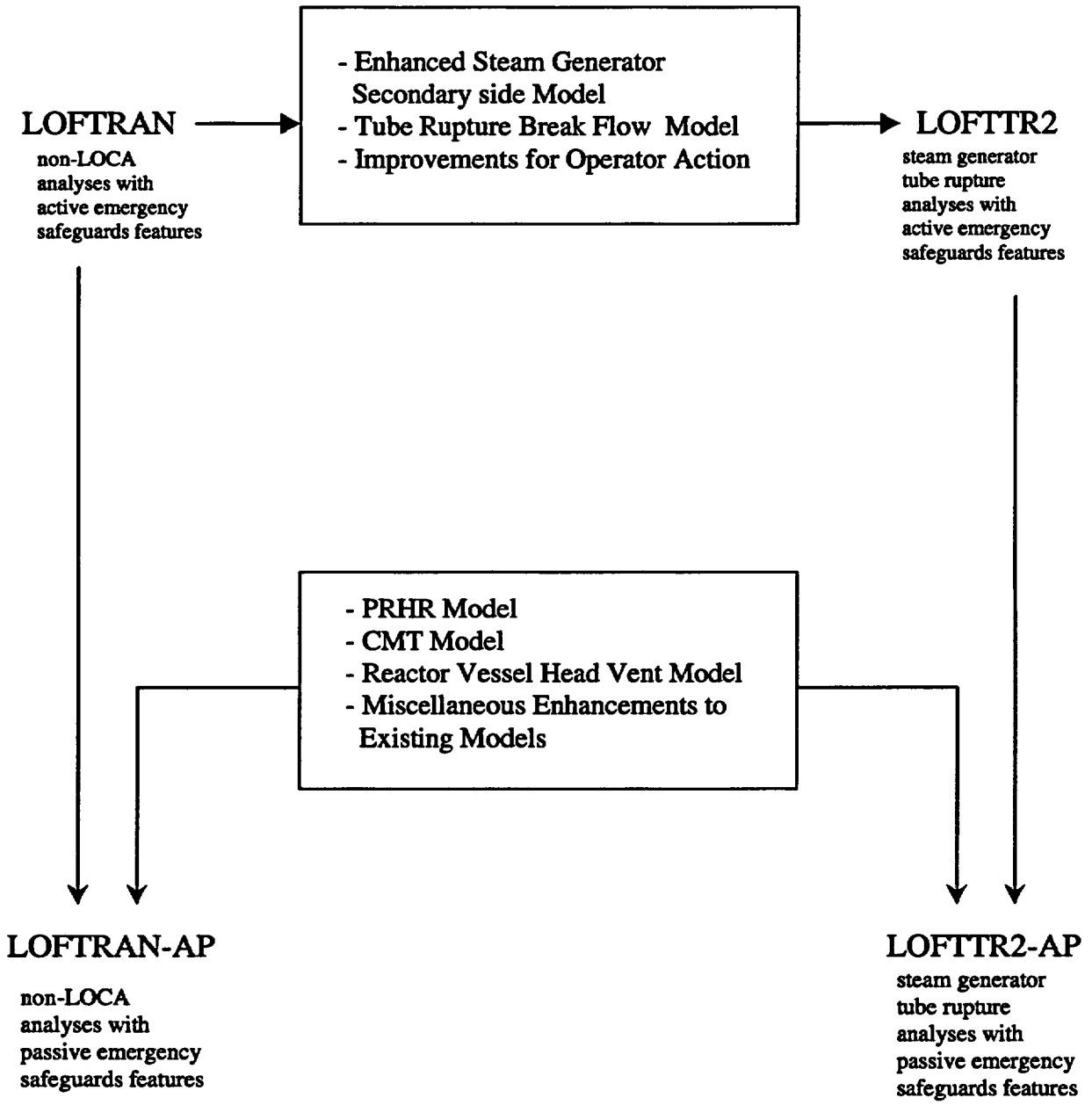

Figure 4-1 Relationship of LOFTRAN Code Versions

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 over-predicting 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

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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

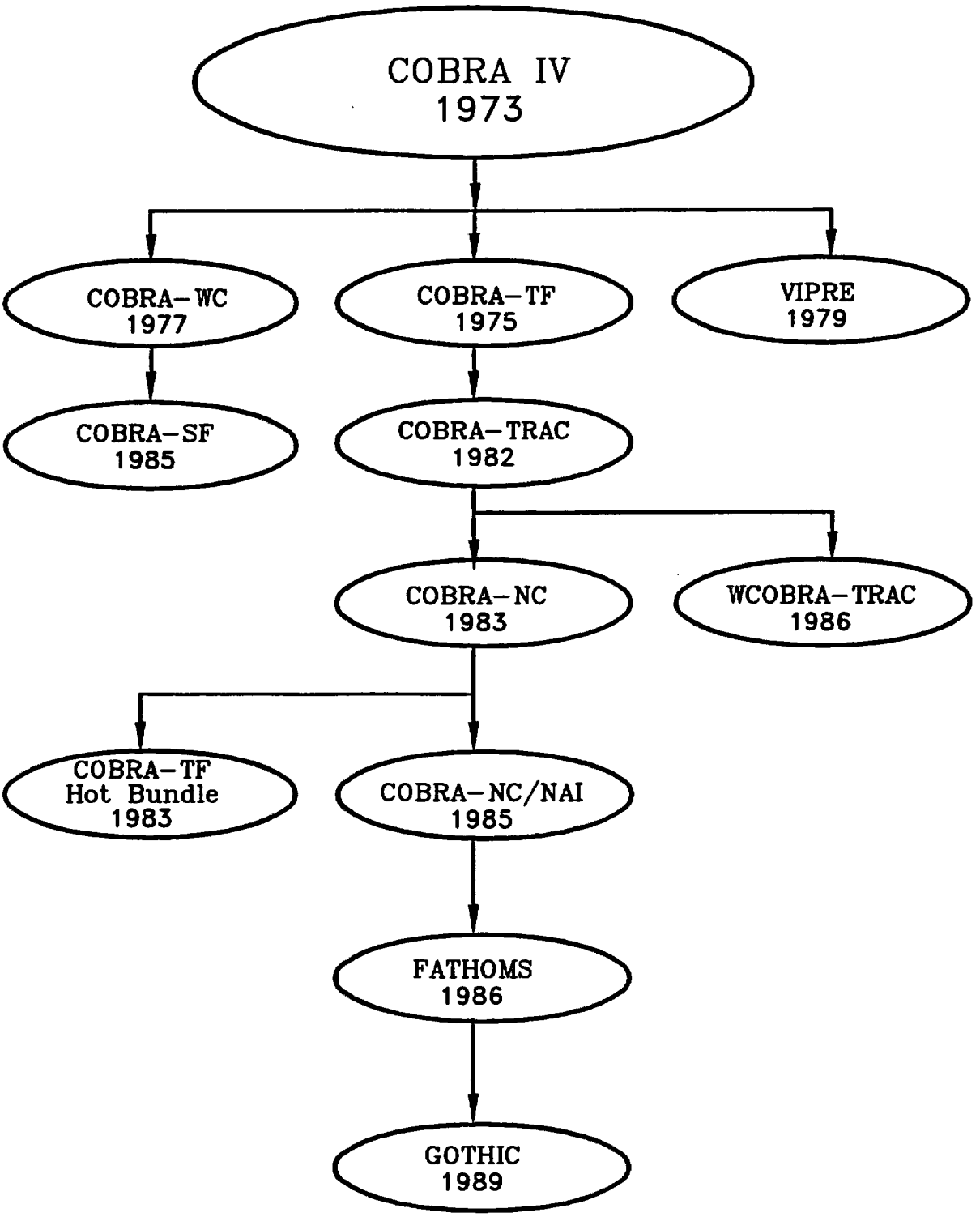

Figure 5-1 Summary of GOTHIC Historical Development

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

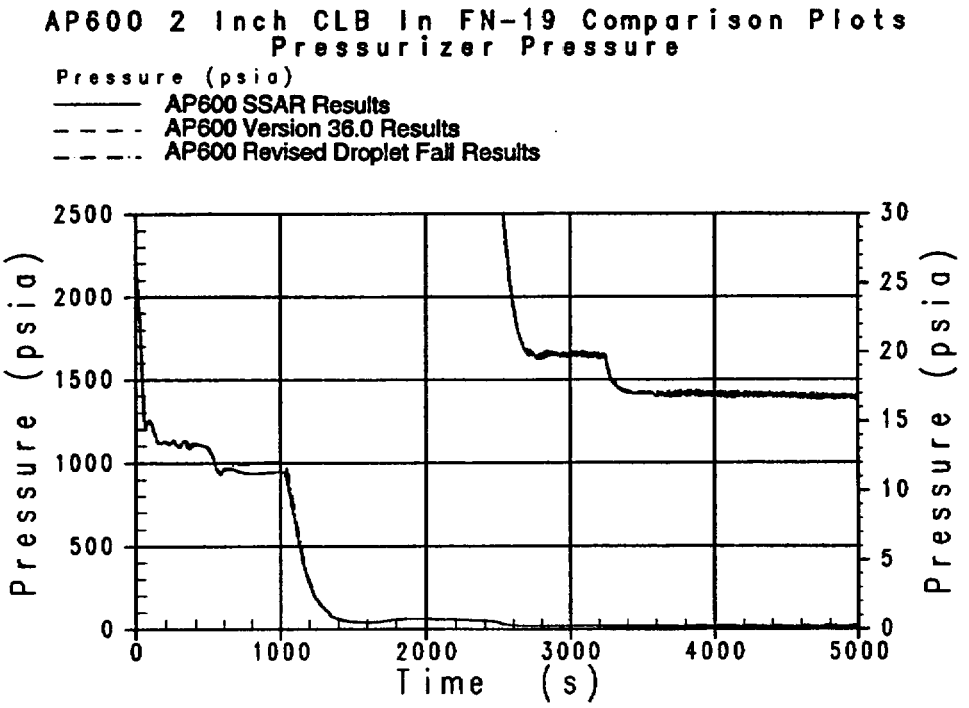

Figure C-1 Pressurizer Pressure

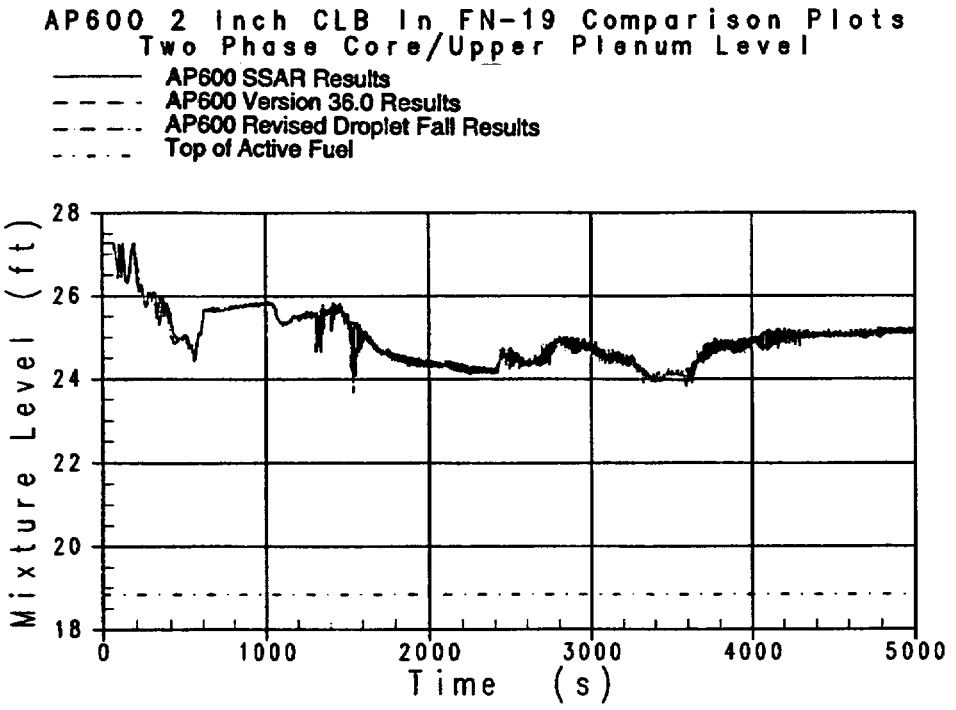

Figure C-2 Core/Upper Plenum Mixture Level

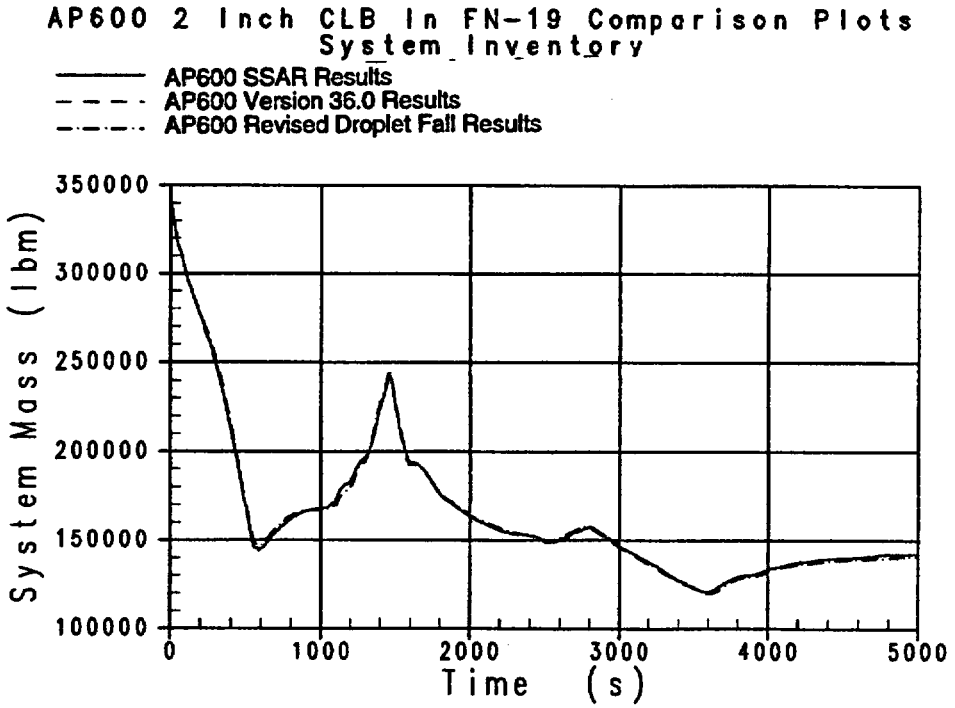

Figure C-3 RCS System Inventory

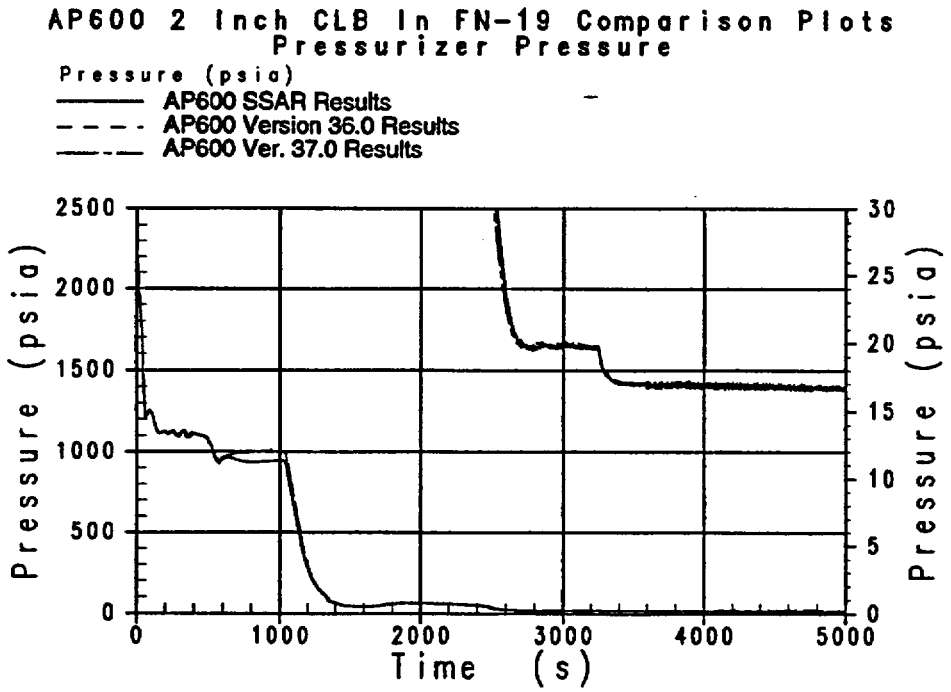

Figure C-4 Pressurizer Pressure

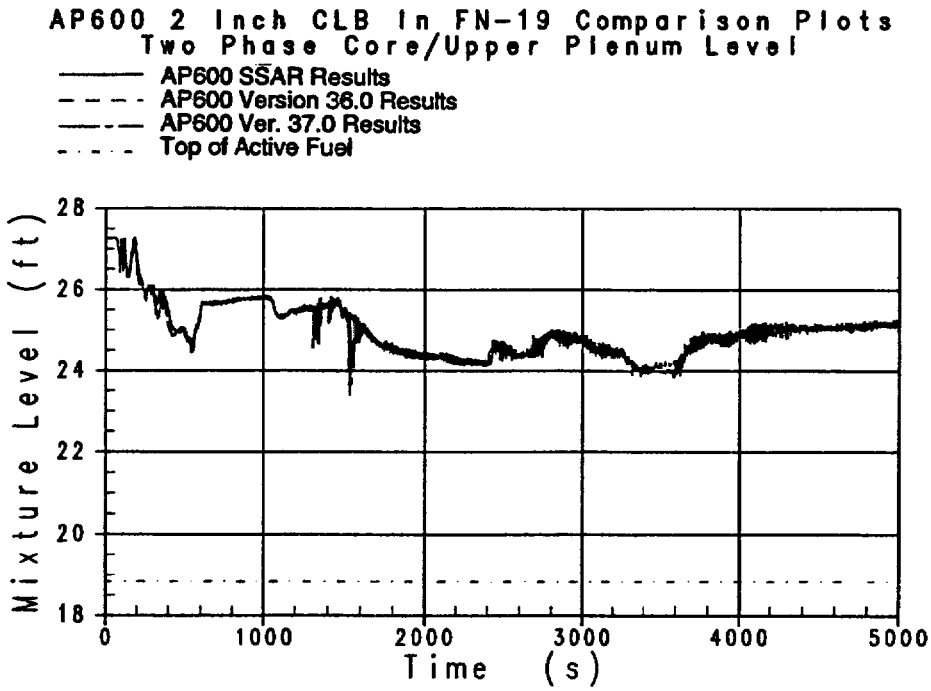

Figure C-5 Core/Upper Plenum Mixture Level

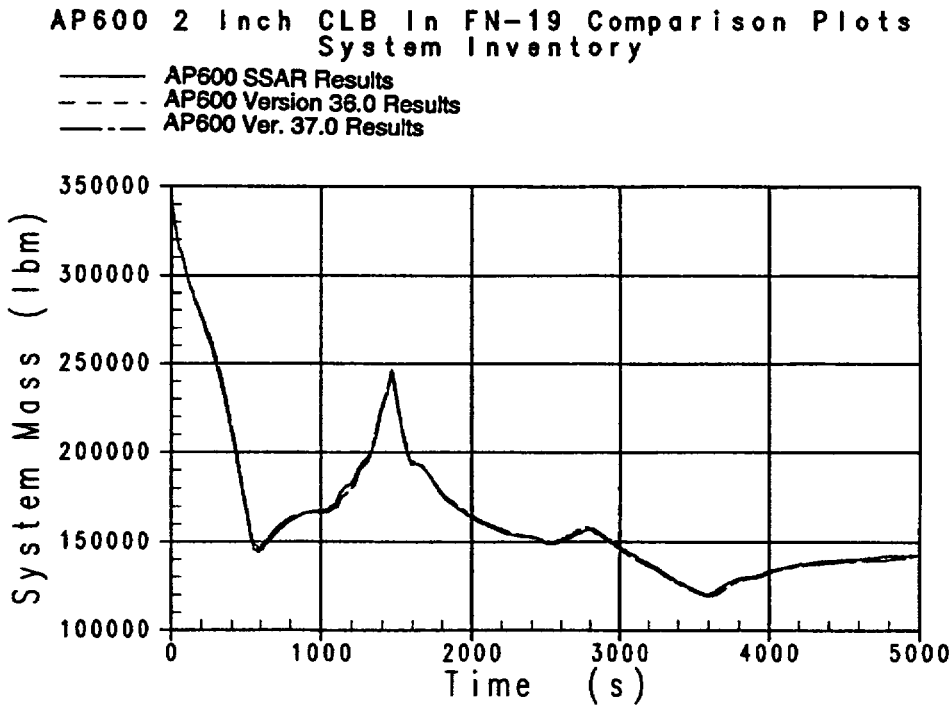

Figure C-6 RCS System Inventory

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

$$Q_{tot} = [\sum q_i] * N_{tubes} \tag{23}$$

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.

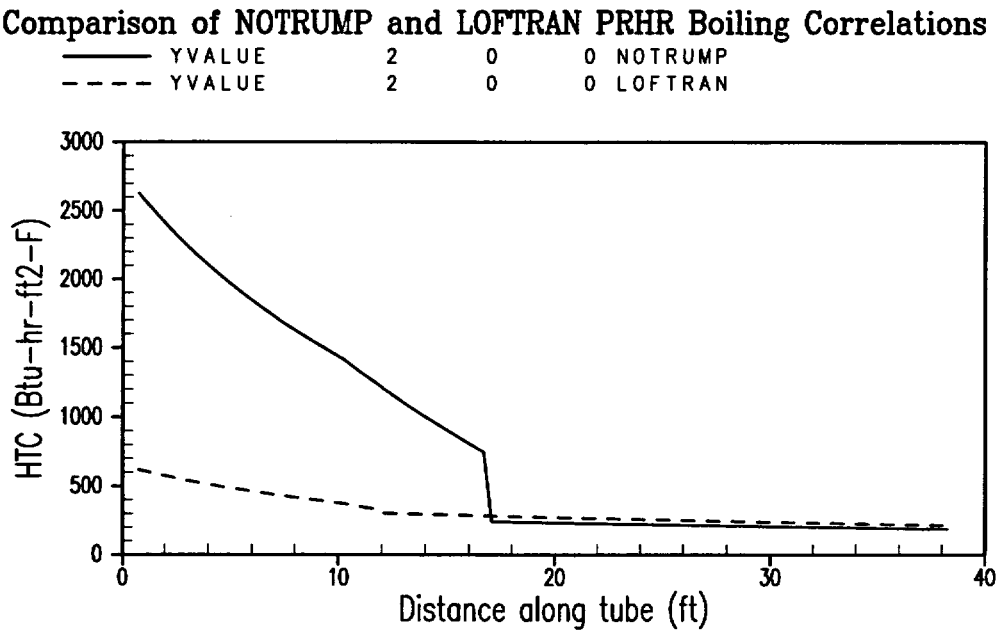

Figure D-1 Heat Transfer Coefficients Versus Distance Along Tube – Single Phase

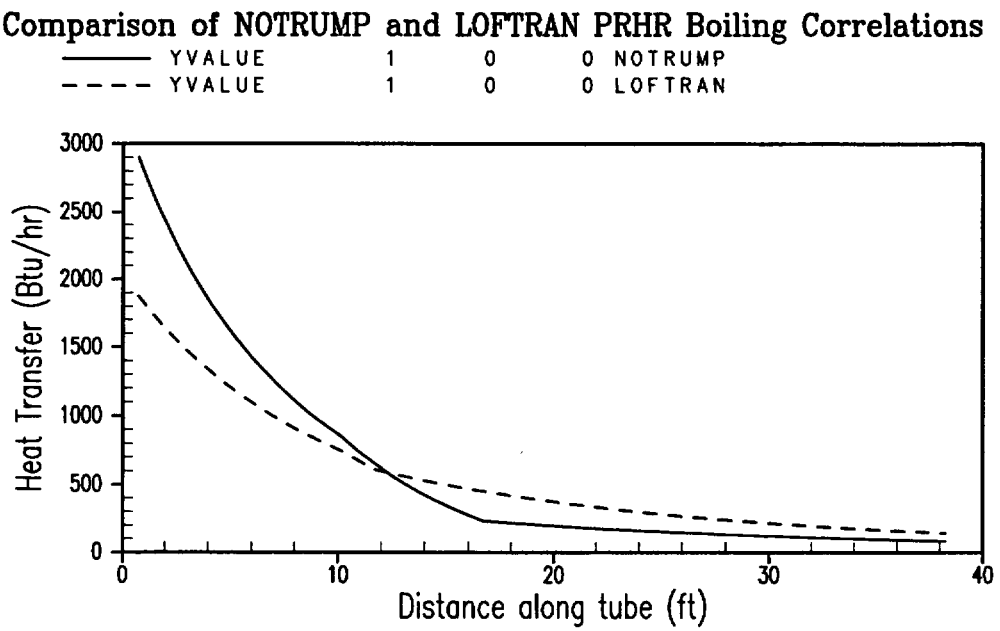

Figure D-2 Heat Transfer Rate Versus Distance Along Tube – Single Phase

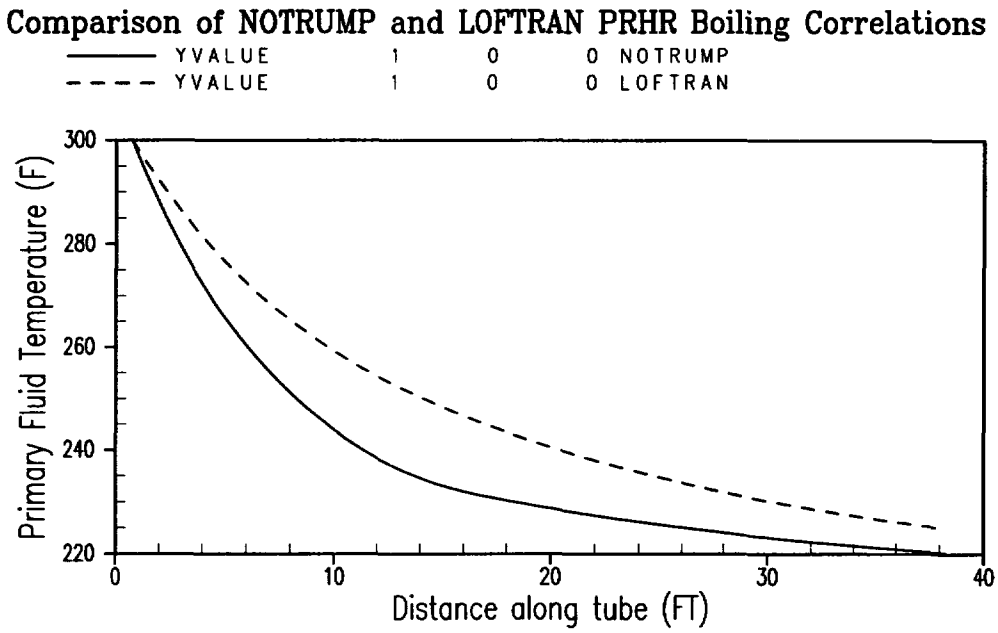

Figure D-3 Primary Fluid Temperature Versus Distance Along Tube – Single Phase

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, Revison 3, AP600 PRHR Heat Exchanger Final Test Report, April 1997.

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

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

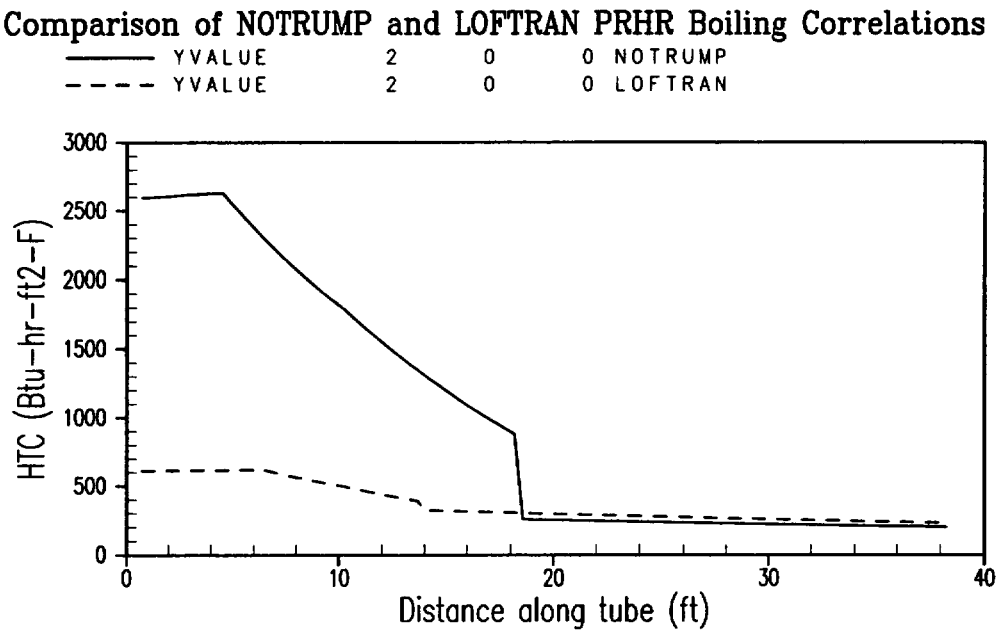

Figure D-4 Heat Transfer Coefficients Versus Distance Along Tube – Two Phase

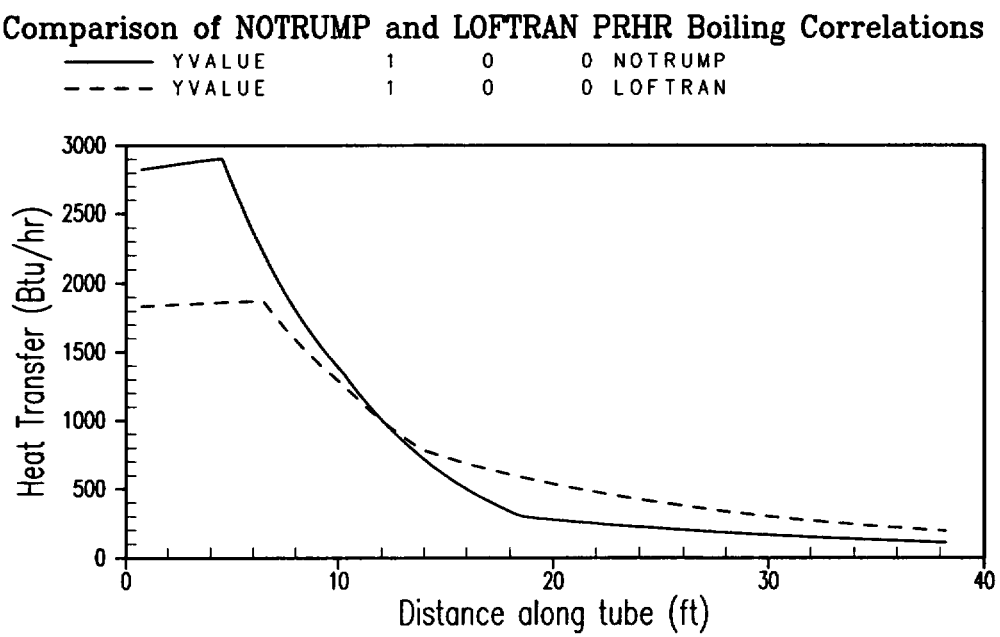

Figure D-5 Heat Transfer Rate Versus Distance Along Tube – Two Phase

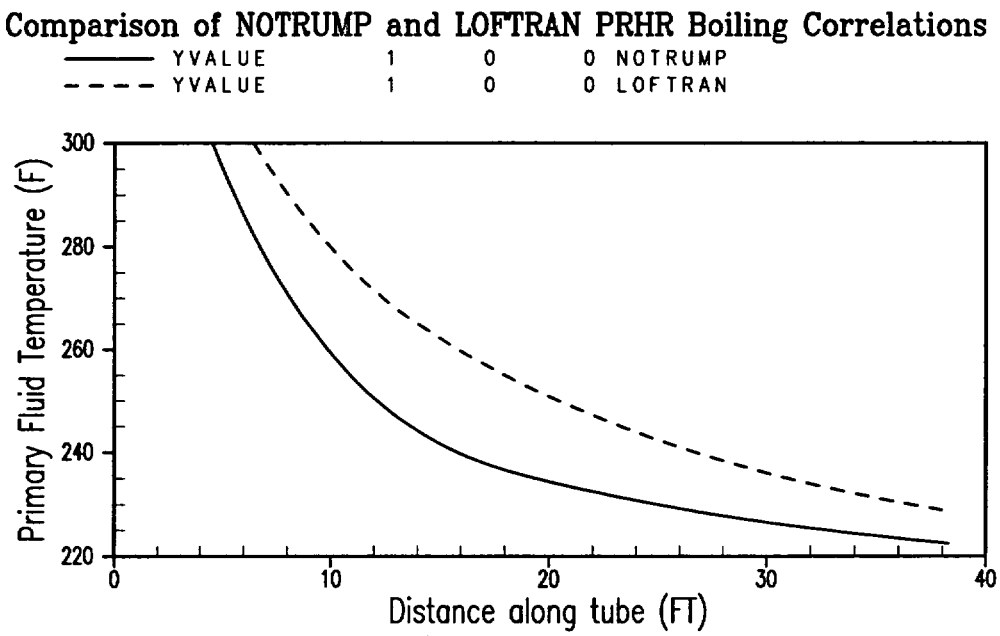

Figure D-6 Primary Fluid Temperature Versus Distance Along Tube – Two Phase

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 [ ]^{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 under-prediction 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 under-predicts 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 over-predicted 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 over-predicted 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) over-predicts 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 over-predicted 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 under-prediction 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 under-predicts 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 over-predicted 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) over-predicts 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 over-predicted 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.

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

E1. 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.

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.

E5. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 1, Section 6, August 1998.

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

E7. WCAP-14807, Revision 5, NOTRUMP Final Validation Report for AP600, Volume 3, RAI 440-721(d), August 1998.

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

APPENDIX F

HOT LEG/UPPER PLENUM ENTRAINMENT SENSITIVITY STUDIES

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-1 with the noding modifications performed shown in Figure F-2.

- 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-3 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-4 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-5 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-6) is also unchanged.

As can be observed from Figure F-7, 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-8 and Figure F-9 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-10) is also approximately the same.

Figure F-11 presents the Pressurizer level response for the two cases. As can be seen, the responses are nearly the same for both cases.

Figure F-12 and Figure F-13 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-14 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-15). 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-16. As expected, the initial ADS-4 liquid discharge is much higher (Figure F-17) until the inventory which resided in the upper plenum and hot leg regions was depleted (Figure F-18). The net effect is a decrease in the ADS-4 vapor discharge rate (Figure F-19) 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-20) 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-21). The net effect of the sensitivity case is that the vessel inventory is substantially decreased over the base model simulation (Figure F-22); however, this inventory is sufficient for adequate core cooling because the ADS-4 continually draws liquid flow through the core (Figure F-17). Even though there is no liquid storage in the upper plenum for the homogenous case (Figure F-23), the coverage percentage (Figure F-24) is not impacted significantly.

The pressurizer mixture level response (Figure F-25) reflects the change in pressure response (Figure F-14) 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.

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.

a,c

Figure F-1 AP1000 NOTRUMP Noding Diagram

a.c

Figure F-2 Vessel Noding Modifications

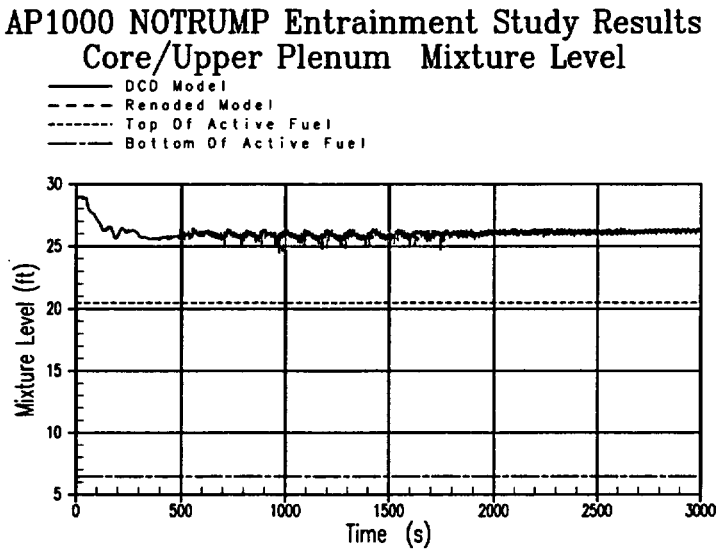

Figure F-3 Core/Upper Plenum Mixture Level Comparison

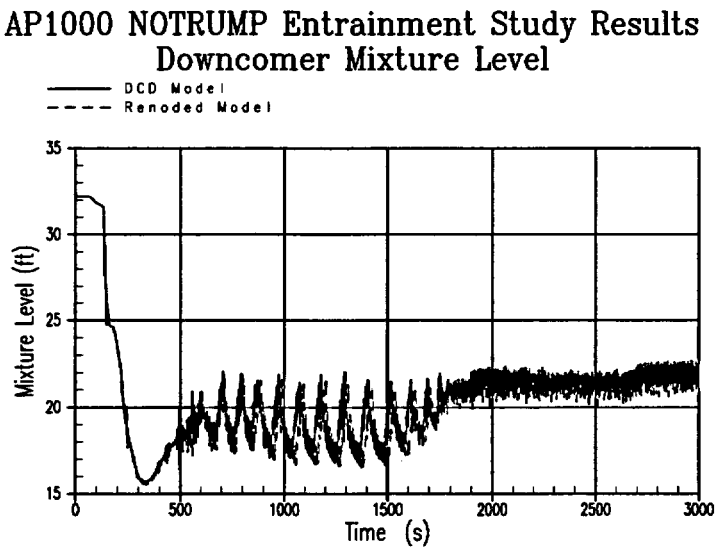

Figure F-4 Downcomer Level Comparison

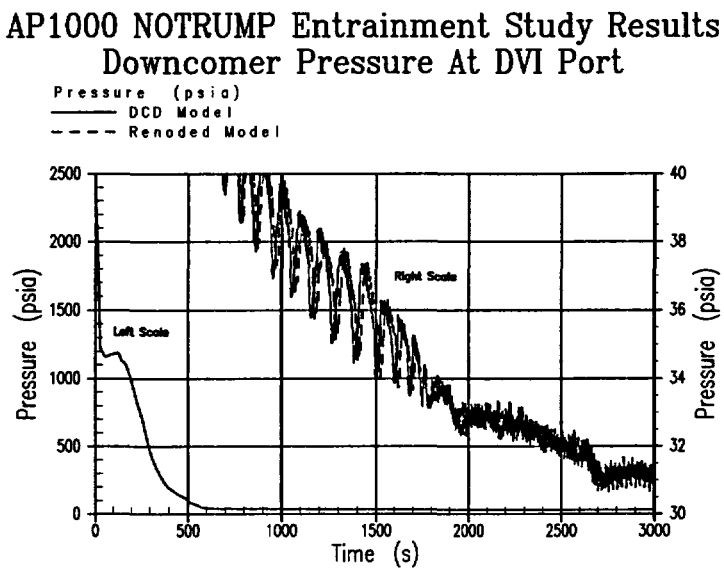

Figure F-5 Downcomer Pressure Comparison

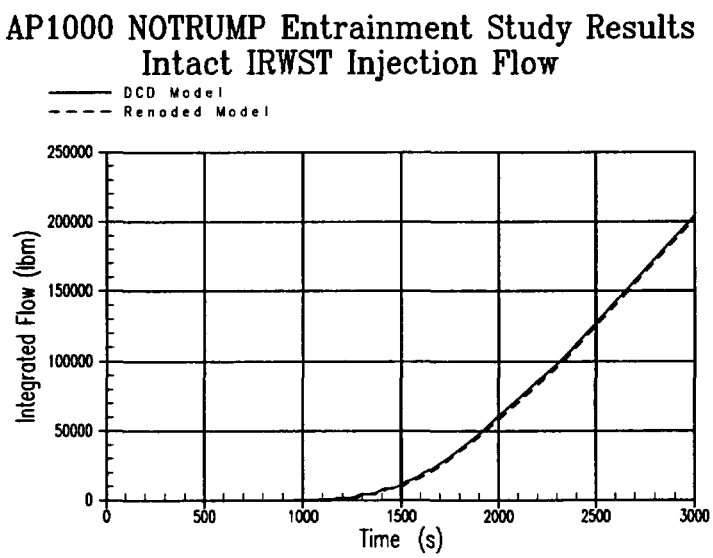

Figure F-6 Intact IRWST Injection Comparison

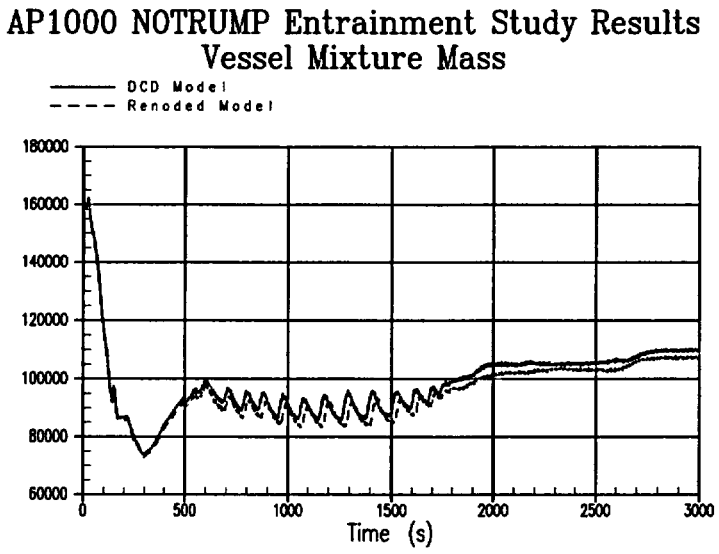

Figure F-7 Vessel Mixture Mass Comparison

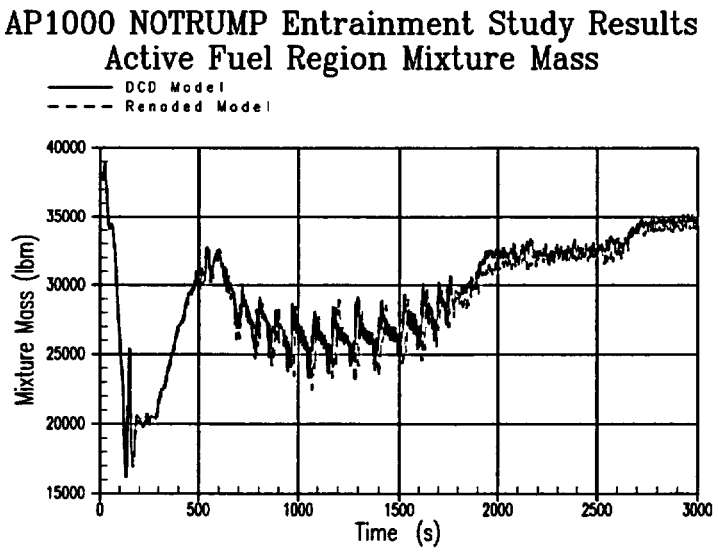

Figure F-8 Active Fuel Region Mixture Mass Comparison

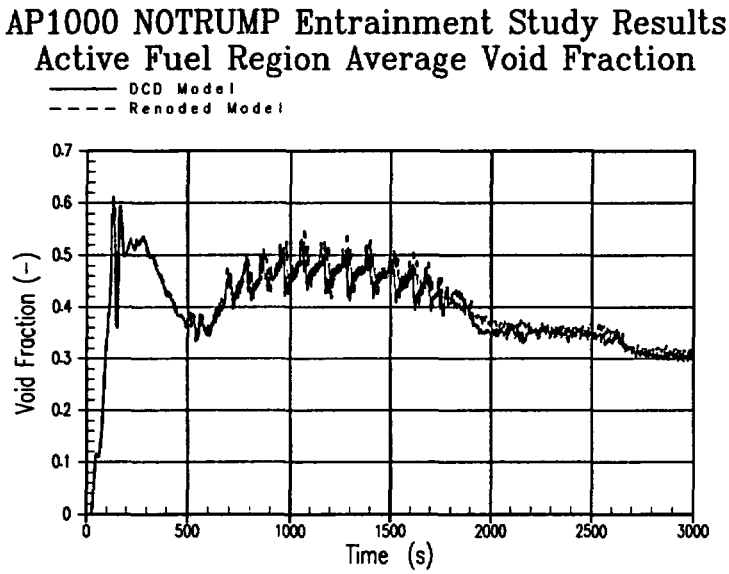

Figure F-9 Active Fuel Region Core Average Void Fraction

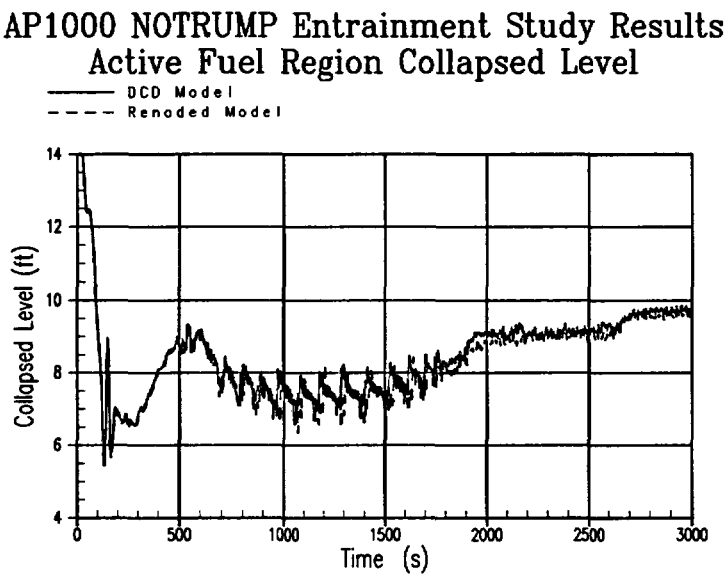

Figure F-10 Active Fuel Region Core Collapsed Level

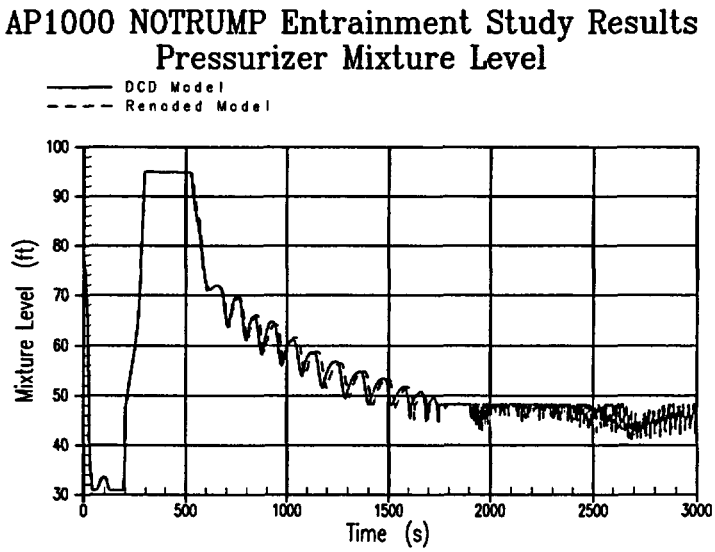

Figure F-11 Pressurizer Level Comparison

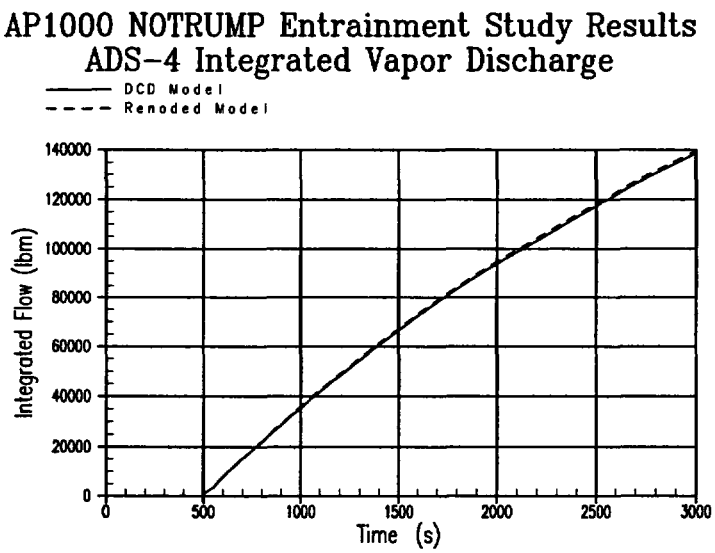

Figure F-12 ADS-4 Liquid Discharge Comparison

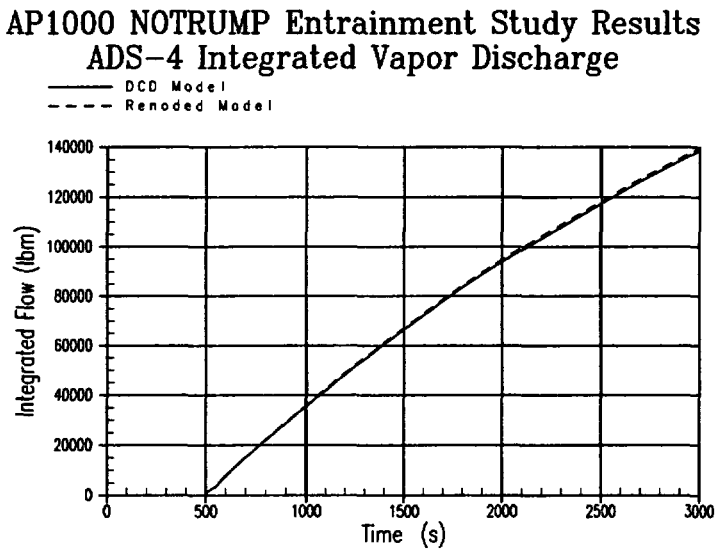

Figure F-13 ADS-4 Vapor Discharge Comparison

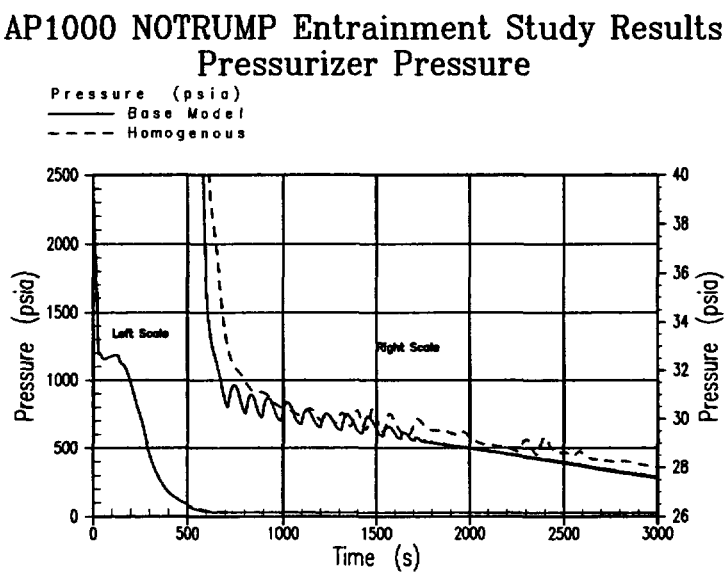

Figure F-14 Downcomer Pressure Comparison

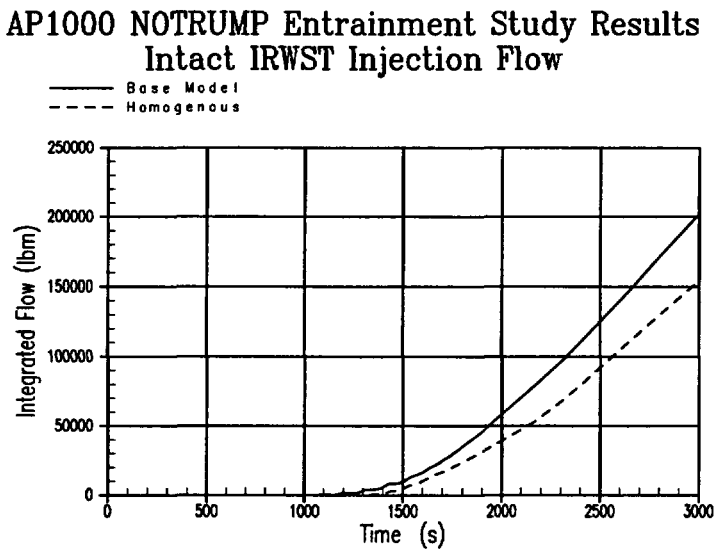

Figure F-15 Intact IRWST Injection Flow

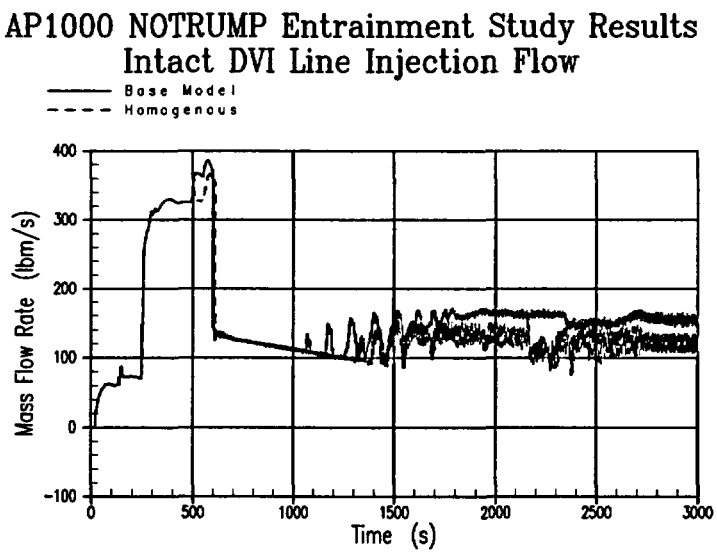

Figure F-16 Intact DVI Line Injection Flow

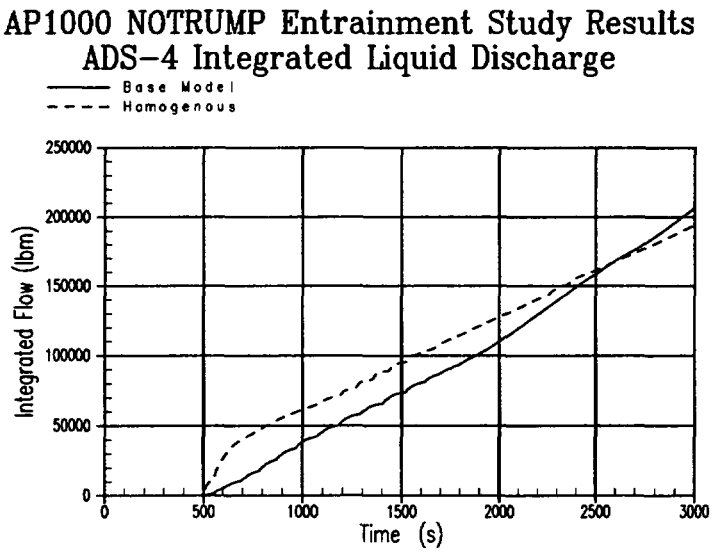

Figure F-17 ADS-4 Integrated Liquid Discharge Comparison

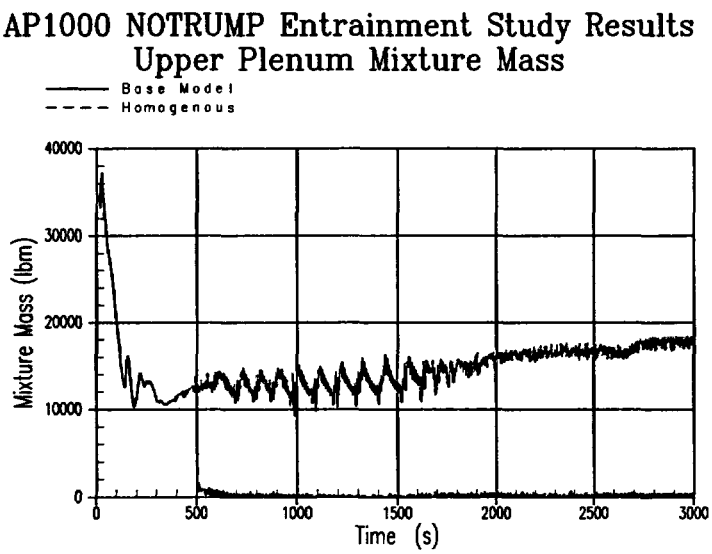

Figure F-18 Upper Plenum Mixture Mass Comparison

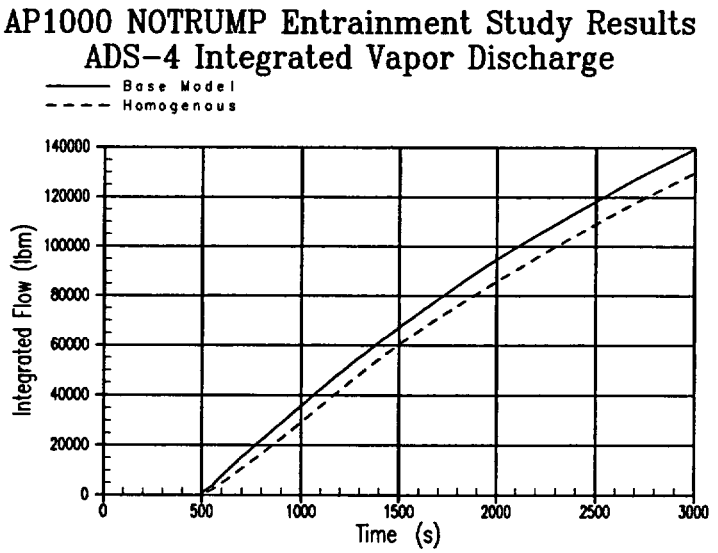

Figure F-19 ADS-4 Integrated Vapor Discharge Comparison

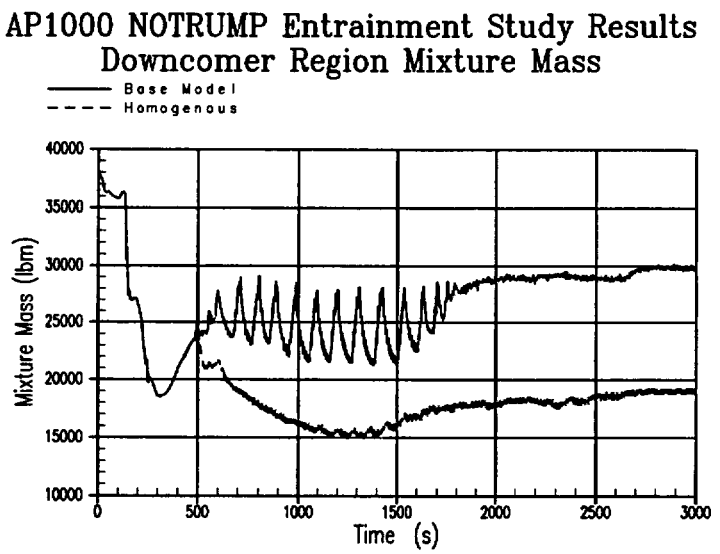

Figure F-20 Downcomer Region Mass Comparison

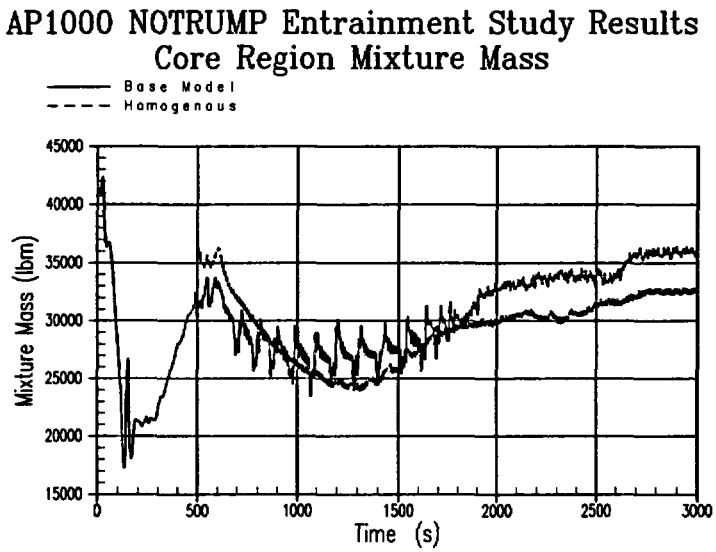

Figure F-21 Core Region Mass Comparison

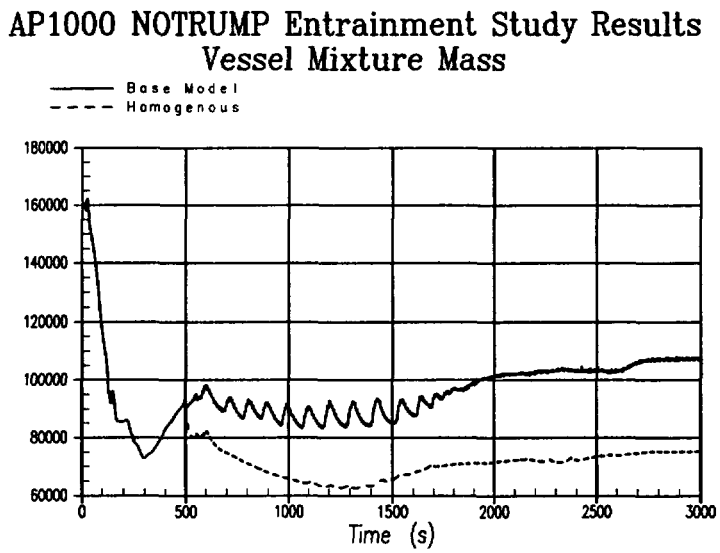

Figure F-22 Vessel Mixture Mass Comparison

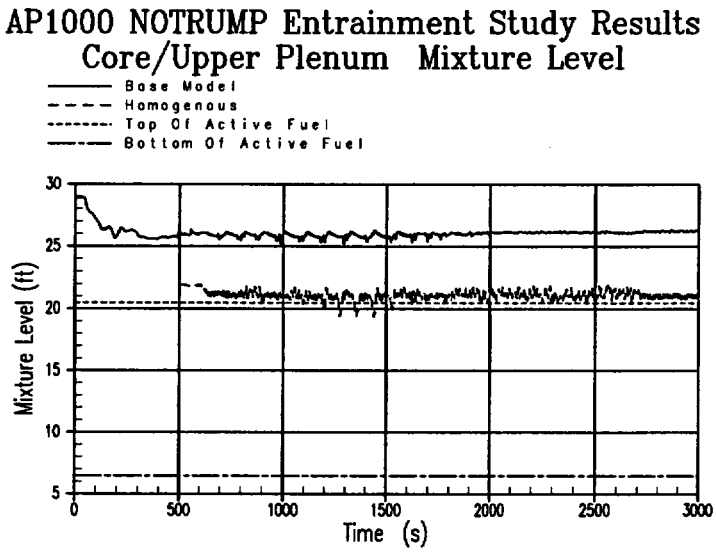

Figure F-23 Core/Upper Plenum Mixture Level Comparison

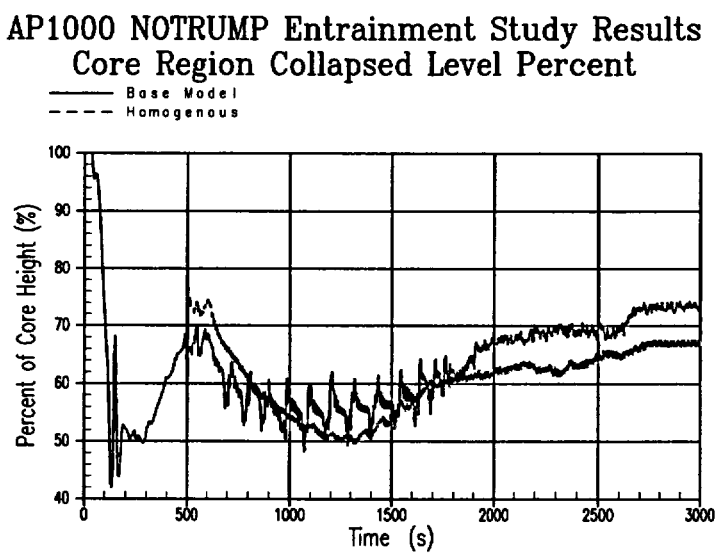

Figure F-24 Core Coverage Percentage Comparison

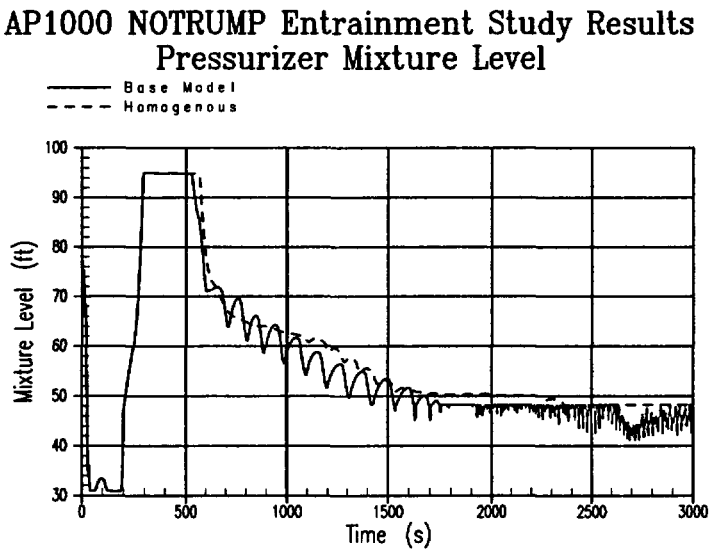

Figure F-25 Pressurizer Mixture Level Comparison

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 ADS4/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:

$$a(z) = 0.925\left(\frac{\rho_g}{\rho_l}\right)^{0.239}\left(\frac{j_g(z)}{V_{bcr}}\right)^{b}\left(\frac{j_g(z)}{j_g(z)+j_l(z)}\right)^{0.6}$$

where:

$$V_{bcr} = \frac{2}{3}(gR_{bcr})^{0.5}$$

$$R_{bcrr} = \left(\frac{1.53}{2/3}\right)^2 \left(\frac{g\sigma}{\rho_l}\right)^{0.25}$$

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.

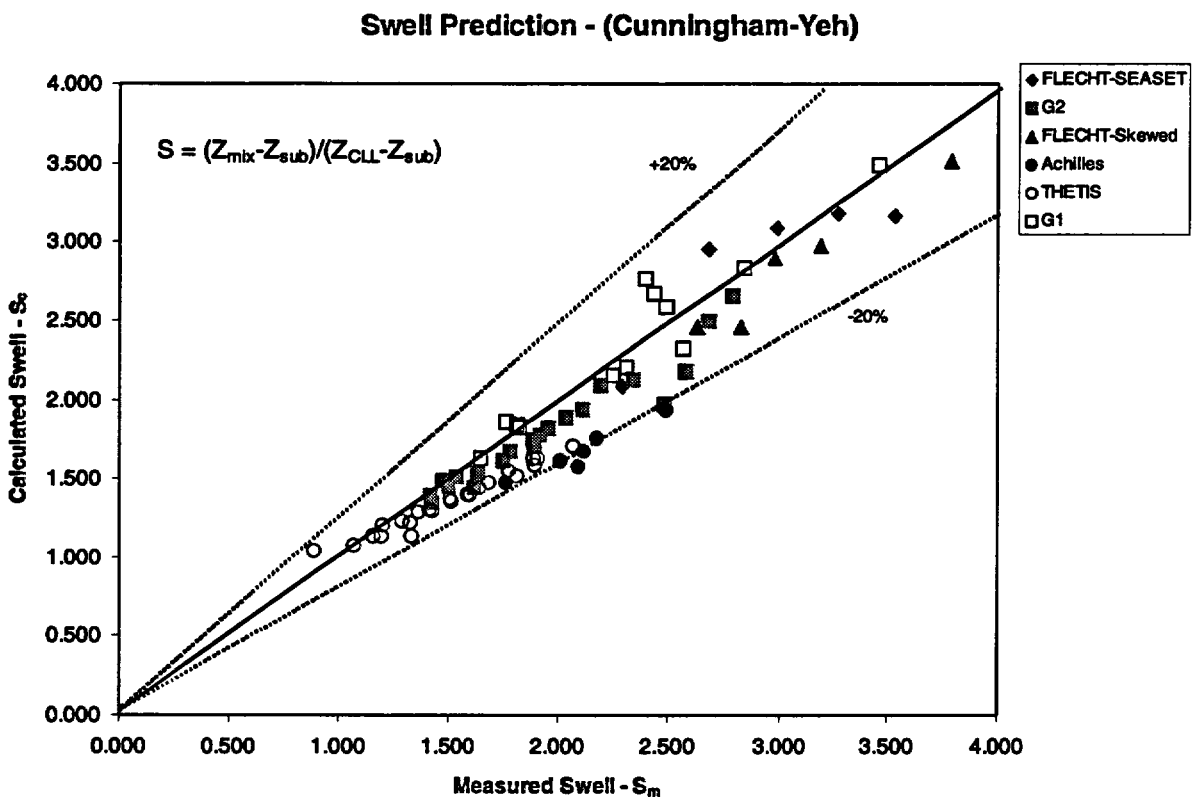

Figure G-1 Calculated Versus Predicted Swell

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.