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May 4, 1995

MFN 070-95
Docket STN 52-004

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

Attention: Theodore E. Quay, Director
Standardization Project Directorate

Subject: ***SBWR Test and Analysis Program Description*, NEDO-32391, Revision B**

Reference: 1) Letter MFN No. 045-95, J. E. Quinn (GE) to R. W. Borchardt (NRC), SBWR Test and Analysis Program Description, NEDO-32391, Revision B, dated April 13, 1995.

This letter transmits 15 additional copies of Revision B of the *SBWR Test and Analysis Program Description* (TAPD) report, NEDO-32391, originally transmitted by Reference 1. These are transmitted for your convenience and help in conducting your review.

This report provides a comprehensive, integrated plan that addresses the testing and analysis elements needed for analysis of the SBWR performance. The changes from Revision A to Revision B are identified by sidebars in the left-hand margins of the affected pages.

Sincerely,

James E. Quinn, Projects Manager
LMR and SBWR Programs

Enclosure: *SBWR Test and Analysis Program Description (TAPD)*, NEDO-32391, Revision B

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NEDO-32391
Revision B
DRF A70-00002
Class 1
April 1995

Licensing Topical Report

SBWR Test and Analysis Program Description

SBWR

9504180148

Welcome to the electronic copy of the SBWR Test and Analysis Program Description and Adobe Acrobat Reader 2.0 for Windows.

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NEDO-32391
Revision B
DRF A70-00002
Class 3
April, 1995

SBWR Test and Analysis Program Description

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April 13, 1995

MFN 045-95
Docket STN 52-004

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

Attention: Richard W. Borchardt, Director
Standardization Project Directorate

Subject: ***SBWR Test and Analysis Program Description*, NEDO-32391, Revision B**

Reference:
1) MFN 018-95, J. E. Quinn (GE) to R. W. Borchardt (NRC), Approach to Achieve Closure of Items Related to the GE SBWR TAPD, dated February 14, 1995.
2) Memorandum to John T. Larkins (ACRS from Dennis M. Crutchfield (NRC), "Draft Safety Evaluation Report (SER) on the Adequacy of the Technical Approach to the Testing and Analysis Program (TAP) for the Simplified Boiling Water Reactor (SBWR) Design", dated November 25, 1994.
3) ACRS Thermal Hydraulic Phenomena Subcommittee Meetings December 15 and 16, 1994, and January 10, 1995.
4) NRC/GE TAPD DSER Meeting January 9, 1995.
5) 417th ACRS Meeting January 12, 1995.

This letter transmits Revision B of the *SBWR Test and Analysis Program Description* (TAPD) report, NEDO-32391, for your review. This report provides a comprehensive, integrated plan that addresses the testing and analysis elements needed for analysis of the SBWR performance. In particular, this revision of the document describes the resolution of testing related issues identified in Reference 1, which summarizes issues raised in the Draft Safety Evaluation Report (Reference 2) and the References 3, 4 and 5 meetings between GE-NE, the NRC and the ACRS.

GE Nuclear Energy

The changes from Revision A to Revision B are identified by sidebars in the left-hand margins of the affected pages.

Sincerely,

RHBuchholz for

James E. Quinn, Projects Manager
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Enclosure: *SBWR Test and Analysis Program Description (TAPD),* NEDO-32391, Revision B

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IDENTIFICATION OF CHANGES

The changes from the previous revision of this document revision are identified by sidebars in the left-hand margins of the affected pages.

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ABBREVIATIONS AND ACRONYMS

ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
ADS	Automatic Depressurization System
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATLAS	GE's 8.6 MW Heat Transfer Loop
ATWS	Anticipated Transients Without Scram
Bldn	Blowdown
BO	Boiloff
BWR	Boiling Water Reactor
CACS	Containment Atmospheric Control System
CCFL	Counter Current Flow Limiting
CISE	Centro Informazioni Studi Esperienze
COL	Combined Operating License
CPR	Critical Power Ratio
CRD	Control Rod Drive
CTP	Core Thermal Power
CRIEPI	Central Research Institute of Electric Power Industry
CSAU	Code Scaling, Applicability and Uncertainty
CSHT	Core Spray Heat Transfer
DBA	Design Basis Accident
DC	Downcomer
DPV	Depressurization Valve
DW, D/W	Drywell
EBWR	Experimental Boiling Water Reactor
ECCS	Emergency Core Cooling System
EOPs	Emergency Operating Procedures
FAPCS	Fuel and Auxiliary Pool Cooling System
FIST	BWR Full Integral Simulation Test
FIX	Swedish Test Loop Used for Testing External Pump Circulation
FMCRD	Fine Motion Control Rod Drive
FRIGG	Research Heat Transfer Loop Operated for Danish Atomic Energy Commission
FW	Feedwater
FWCS	Feedwater Control System
GDCS	Gravity-Driven Cooling System
GE	General Electric Company

ABBREVIATIONS AND ACRONYMS (Continued)

GEXL	General Electric Critical Quality Boiling Length Correlation
GIRAFFE	Gravity-Driven Integral Full-Height Test for Passive Heat Removal
GIST	GDCS Integral System Test
HCU	Hydraulic Control Unit
HVAC	Heating, Ventilating and Air Conditioning
IC	Isolation Condenser
ICS	Isolation Condenser System
INEL	Idaho National Engineering Laboratory
LASL	Los Alamos Scientific Laboratory
LB	Large Break
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPCI	Low Pressure Coolant Injection
MCPR	Minimum Critical Power Ratio
MIT	Massachusetts Institute of Technology
MPL	Master Parts List
MSIV	Main Steamline Isolation Valve
MSL	Main Steamline
MW	Megawatt
NBS	Nuclear Boiler System
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
P&ID	Process and Information Diagram
PANDA	Passive Nachwarmeabfuehr-und Drueckabbau-Testanlage (Passive Decay Heat Removal and Depressurization Test Facility)
PANTHERS	Performance Analysis and Testing of Heat Removal Systems
PAR	Passive Autocatalytic Recombiners
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PIRT	Phenomena Identification and Ranking Tables
PSTF	Pressure Suppression Test Facility
QDB	Qualification Database
RC&IS	Rod Control and Information System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SB	Small Break

ABBREVIATIONS AND ACRONYMS (Continued)

SBWR	Simplified Boiling Water Reactor
S/C	Suppression Chamber (wetwell)
SDC	Shutdown Cooling
SIET	Societa Informazioni Esperienze Termoidrauliche
SLCS	Standby Liquid Control System
SPERT	Special Power-Excursion Reactor Tests
SRV	Safety/Relief Valve
SSAR	Standard Safety Analysis Report
SSLC	Safety System Logic Control
SSTF	Steam Sector Test Facility
TAPD	Test and Analysis Program Description
TCV	Turbine Control Valve
THTF	Thermal-Hydraulic Test Facility
TLTA	Two-Loop Test Apparatus
TPS	Turbine Protection System
TRAC	Transient Reactor Analysis Code
TRACG	Transient Reactor Analysis Code, GE version
TT	Turbine Trip
UCB	University of California, Berkeley
VB	Vacuum Breaker
WW	Wetwell

1.0 INTRODUCTION

1.1 Purpose

The purpose of the Simplified Boiling Water Reactor (SBWR) Test and Analysis Program Description (TAPD) is to provide, in one document, a comprehensive, integrated plan that addresses the testing and analysis elements needed for analysis of SBWR steady-state and transient performance. The program was developed by:

- Study of the calculated SBWR transients and identification of important phenomena.
- Identification of the unique SBWR design features and their effect on transient performance.
- Systematic definition of experimental and analytical modeling needs.
- Evaluation of the current experimental and analytical model plan against these needs.
- Definition of modifications as necessary.

This document describes the steps in this process leading to the final Test and Analysis Plan (Appendix A). The TRACG computer code is used for the analysis of SBWR transients, Loss-Of-Coolant Accidents (LOCAs), Anticipated Transients Without Scram (ATWS) and stability. The Test Plan has been cross-referenced against the identified phenomena to create the TRACG Qualification Matrix. Section 1.3 describes in more detail the strategy employed to arrive at these objectives. The use of specific tests in the development of TRACG models, for test predictions and for post-test validation, is addressed in this report. Descriptions of the SBWR-specific test facilities and their fidelity with respect to scaling the SBWR plant are provided in Appendices A and B.

The SBWR TAPD thus provides the technology basis for determining the performance of the plant for transients and accidents. It ties together the ongoing diverse experimental and analytical efforts in support of SBWR certification. The ultimate output from this activity is a set of validated analytical methods (primarily the TRACG computer code) for SBWR performance analysis.

1.1.1 Scope

The SBWR Test and Analysis Program Description is directed at providing a sound technology basis for the prediction of SBWR system performance during normal operation, transients and LOCAs. The document scope includes (1) steady-state operation and startup conditions, (2) transients and ATWS, (3) stability, and (4) LOCA. LOCA response covers the vessel response [levels and peak cladding temperature (PCT)] with operation of the Emergency Core Cooling Systems (ECCS), as well as the containment pressure and temperature response to postulated breaks. Long-term core cooling by inventory makeup is also considered.

The document does not address "severe accident" issues. The requirement to design the containment to handle hydrogen generation assuming 100% metal-water reaction is, however, addressed as a Design Basis requirement. Issues related not to thermal-hydraulics but, for example, to material properties, crack resistance, water chemistry, etc., are not covered in this plan.

The TAPD focus is illustrated in Figure 1.1-1. Transients and accidents, short of severe core damage, have been analyzed and the experimental and modeling needs incorporated into the plan. In the time domain, the focus of the studies has been on the first three days following a postulated accident or transient. Quasi-steady-state conditions prevail well before this point in time. Interactions with active systems such as the Fuel and Auxiliary Pool Cooling System (FAPCS) have been studied. No new phenomena are introduced beyond this point.

The experimental and analytical modeling needs were analyzed in the context of the applicable criteria of 10CFR52.47(b)(2)(i)(A), which require in part that:

- The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.
- Interdependent effects among the safety features of the design have been found to be acceptable by analysis, appropriate test programs, experience, or a combination thereof.
- Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

The term "safety feature" in the preceding paragraph is understood to include safety-related passive systems as well as other active systems which may be available to operators during accidents or transients. The Bottom-Up process described in Section 3 specifically examines all SBWR-unique features that are relevant to safety. Issues related to these features have been evaluated and the supporting technology basis (analysis, experimental data, plant data) documented. Interdependent effects among safety features have been specifically considered. Analyses have been performed (Appendix C) to screen interactions that deserve experimental validation. Finally, a test program has been established which provides a sufficient database for the qualification of the TRACG Code for SBWR safety analysis.

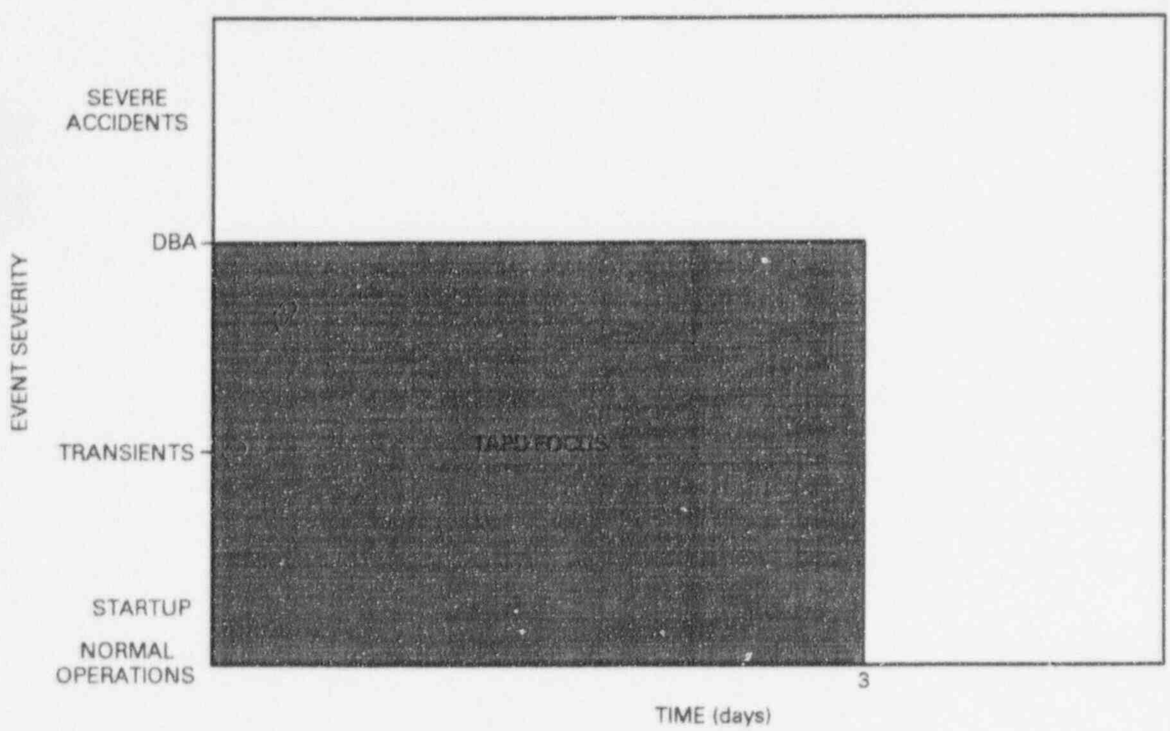


Figure 1.1-1 TAPD Focus

1.2 Background

SBWR Design Evolution:

The SBWR design is an evolutionary step in boiling water reactor (BWR) design which traces its commercial demonstration and operating plant history back before 1960 (Figure 1.2-1). Since its inception, the BWR has had plant simplification as a goal for each product improvement (Figure 1.2-2). The SBWR has major simplifying improvements drawn from predecessor designs, notably pressure-suppression containment, natural circulation, isolation condenser handling of waste heat, and gravity-driven makeup water systems (Table 1.2-1). The incorporation of these features from predecessor designs into the SBWR has emphasized employment of passive means of dealing with operational transients and hypothetical LOCAs. The result of this evolution of previously licensed plant features is simplified operator response to these events (most plant upset conditions are dealt with in the same manner, as typified by the hypothetical steamline break), and a lengthened operator response time for all hypothetical events (from minutes for previously licensed reactors to days for the SBWR). Most features of the SBWR have been taken directly from licensed commercial BWRs and reviewed and redesigned as appropriate for the SBWR (Table 1.2-2). The SBWR draws together the best of previously licensed plant features to continue the simplification process. As an example, the evolution of the containment is shown in Figure 1.2-3.

Analysis and Design Tools:

As implied above, data available from operating plants and from the testing and licensing efforts done to license the predecessor designs (most recently, ABWR) is the principal foundation of SBWR technology. As a measure of the SBWR's reliance on demonstrated technology, approximately 50% of the content of the SBWR SSAR is technically identical or technically similar (with minor differences) to the ABWR SSAR [31]. The 930 reactor-year database [40] of feature performance in operating reactors, combined with the recent thorough licensing review of the ABWR (Final Design Approval received July 1994), provides well-qualified foundation from which to make the modest extrapolations to the SBWR.

To make that extrapolation, GE has developed one computer code (TRACG) to use for design and for three out of the four most limiting licensing analyses. The TRACG Code, validated by operating plant experience and appropriate testing, is used to analyze the challenges to the fuel (10CFR50.46 and Appendix K, SSAR Section 6.3), the challenges to the containment (SSAR Section 6.2), and many of the operational transients (MCPR, SSAR Chapter 15). The radiological responses to hypothetical accidents are also presented in SSAR Chapter 15, but do not use TRACG for analysis. Thus, TRACG draws from the very large database of licensed BWRs which includes all features of the SBWR (albeit in various configurations) and appropriate testing, and allows direct application to SBWR design and analysis.

1.2.1 Use of TRACG

The TRACG Code and its application to the SBWR is documented in a series of GE Nuclear Energy Topical Reports ([1], [2], and [7]).

TRACG is a GE proprietary version of the Transient Reactor Analysis Code (TRAC). It is a best-estimate code for analysis of BWR transients ranging from simple operational transients to design basis LOCAs, stability, and ATWS.

1.2.1.1 Background

TRAC was originally developed for pressurized water reactor (PWR) analysis by Los Alamos National Laboratory (LANL), the first PWR version of TRAC being TRAC-P1A. The development of a BWR version of TRAC started in 1979 in a close collaboration between GE and Idaho National Engineering Laboratory. The objective of this cooperation was the development of a version of TRAC capable of simulating BWR LOCAs. The main tasks consisted of improving the basic models in TRAC for BWR applications and developing models for the specific BWR components. This work culminated in the mid-eighties with the development of TRACB04 at GE and TRAC-BD1/MOD1 at INEL, which were the first major versions of TRAC having BWR LOCA capability. Due to the joint development effort, these versions were very similar, having virtually identical basic and component models. The GE contributions were jointly funded by GE, the Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI) under the REFILL/REFLOOD and FIST programs.

The development of the BWR version has continued at GE since 1985. The objective of this development was to upgrade the capabilities of the code in the areas of transient, stability and ATWS applications. Major improvements included the implementation of a core kinetics model and addition of an implicit integration scheme into TRAC. The containment models were upgraded for SBWR applications, and the simulation of the fuel bundle was also improved. TRACG was the end result of this development.

1.2.1.2 Scope and Capabilities

TRACG is based on a multi-dimensional two-fluid model for the reactor thermal-hydraulics and a three-dimensional neutron kinetics model.

The two-fluid model used for the thermal-hydraulics solves the conservation equations for mass, momentum and energy for the gas and liquid phases. TRACG does not include any assumptions of thermal or mechanical equilibrium between phases. The gas phase may consist of a mixture of steam and a noncondensable gas, and the liquid phase may contain dissolved boron. The thermal-hydraulic model is a multi-dimensional formulation for the vessel component and a one-dimensional formulation for all other components.

The conservation equations for mass, momentum and energy are closed through an extensive set of basic models consisting of constitutive correlations for shear and heat transfer at the gas/liquid interface as well as at the wall. The constitutive correlations are flow regime dependent and are determined based on a single flow regime map, which is used consistently throughout the code.

In addition to the basic thermal-hydraulic models, TRACG contains a set of component models for components, such as channels, steam separators and dryers. TRACG also contains a control system model capable of simulating the major control systems such as reactor pressure vessel (RPV) pressure and water level.

The neutron kinetics model is consistent with the GE core simulator code PANACEA. It solves a modified one-group diffusion model with six delayed neutron precursor groups. Feedback is provided from the thermal-hydraulic model for moderator density, fuel temperature, boron concentration and control rod position.

The TRACG structure is based on a modular approach. The TRACG thermal-hydraulic model contains a set of basic components, such as pipe, valve, tee, channel, steam separator, heat exchanger and vessel. System simulations are constructed using these components as building blocks. Any number of these components may be combined. The number of components, their interaction, and the detail in each component are specified through code input. TRACG consequently has the capability to simulate a wide range of facilities, ranging from simple separate effects tests to complete plants.

TRACG has been extensively qualified against separate effects tests, component performance data, integral system effects tests and full-scale plant data. A detailed documentation of the qualification is contained in the TRACG qualification report NEDE-32177P [2].

1.2.1.3 Scope of Application of TRACG to SBWR

The TRACG computer code has been qualified to Level 2 status at GE-NE. Thus, the code configuration is controlled, and the models and the results of validation testing have been reviewed and approved by an independent Design Review Team. In the development process, the separate effects and component data were used for model development and refinement.

The total effort and extent of qualification performed on TRACG, since its inception in 1979, now exceeds, both in extent and breadth, that for any other engineering computer program which GE has submitted to the NRC for design application approval. The Level 2 application of TRACG includes LOCA analyses, transients, ATWS and Stability Analyses for the reactor and containment. Table 1.2-3 compares the analytical methods used for ABWR and SBWR analysis. The table shows that GE has taken a major step forward in utilizing one code (TRACG) for the bulk of the safety analysis. This results in greater consistency and simplification of the analysis process. The use of TRACG to unify the LOCA analysis for the reactor vessel and containment is particularly important for the SBWR because the two regions are closely coupled during the transient.

While TRACG is used for all the analyses given in Table 1.2-3, the application of TRACG in the design process is different for ATWS and stability. For LOCA (ECCS and containment) and transient analysis, GE performs SSAR calculations utilizing a best-estimate analytical technique which realistically describes system behavior and appropriately considers uncertainties in the analysis methods and inputs per the requirements of 10CFR50.46(a)(1)(i). The ATWS calculations are performed as best-estimate calculations. For stability analysis, NRC approved methodology (FABLE) is used in the design process for determination of core and channel stability margins. TRACG is used for the evaluation of overall plant stability. TRACG has also been used to study the possibility of oscillations during the plant startup transient.

1.2.1.3.1 Transient Analysis

TRACG is used to perform safety analyses of nearly all of the Anticipated Operational Occurrences (AOO) described in SSAR Chapter 15, and of the ASME reactor vessel overpressure protection events in SSAR Chapter 5. The Loss of Feedwater Heating and the Control Rod Withdrawal Error events presented in SSAR Chapter 15 are analyzed using the GE 3-D core simulator model. Other SSAR Chapter 15 exceptions are the control rod drop and the fuel-handling accidents, and radiological calculations for all postulated accidents.

The analysis determines the most limiting event for the AOOs in terms of Critical Power Ratio (CPR) and margin loss (ΔCPR) and establishes the operating limit minimum CPR (OLMCPR). The OLMCPR includes the statistical CPR adder which accounts for uncertainty in calculated results arising from uncertainties associated with the TRACG model, initial conditions, and input parameters. Sensitivity analysis of important parameters affecting the transient results is performed using TRACG. Concepts derived from the Code Scaling, Applicability, and Uncertainty (CSAU) methodology are utilized for quantifying the uncertainty in calculated results.

The analysis also determines the most limiting overpressure protection events in terms of peak vessel pressure. The results are used to demonstrate adequate pressure margin to the reactor vessel design limit with the SBWR design safety/relief valve capacity. The overpressure protection analysis is performed based on conservative initial conditions and input values.

1.2.1.3.2 ATWS Analysis

TRACG is used for evaluation of the ATWS events in SSAR Chapter 15. The analysis determines the most limiting ATWS events in terms of reactor vessel pressure, heat flux, neutron flux, peak cladding temperature, suppression pool temperature, and containment pressure. The results are used to demonstrate the capability of the SBWR mitigation design features to comply with the ATWS licensing criteria.

1.2.1.3.3 ECCS/LOCA Analysis

TRACG is used for evaluation of the complete spectrum of postulated pipe break sizes and locations, together with possible single active failures, for Section 6.3 of the SBWR SSAR. This evaluation determines the worst case break and single failure combinations. The results are used to demonstrate the SBWR Emergency Core Cooling System (ECCS) capability to comply with the licensing acceptance criteria.

A sensitivity analysis of important parameters affecting LOCA results is performed using TRACG. For the SBWR, the LOCA analysis results are adjusted so that they provide 95% probability LOCA results for use as the licensing basis. The SBWR LOCA results have large margin with respect to the licensing acceptance criteria.

1.2.1.3.4 Containment Analysis

TRACG is also used for evaluation of containment response during a LOCA. The analysis determines the most limiting LOCA for containment (or Design Basis Accident, DBA) in terms of containment pressure and temperature responses. The DBA is determined from consideration of a full spectrum of postulated LOCAs. The results are used to demonstrate compliance with the SBWR containment design limits.

Sensitivity of the containment response to parameters identified as important is evaluated using TRACG to assess the effect of uncertainties of these parameters on the containment responses. The procedure derived from the CSAU methodology (Subsection 1.2.2) is used for this purpose.

1.2.2 Major SBWR Test Facilities

GE has used a procedure similar to the Code Scaling, Applicability and Uncertainty (CSAU) methodology developed by the NRC [4], [6] and submitted to the NRC by GE letter [41]. This procedure developed a list of phenomena important to the SBWR behavior in a large number of anticipated and hypothetical events and matched them against information available from operating plant and/or test experience. The Phenomena Identification and Ranking Table (PIRT) discussed in Section 2 of this report identifies specific governing phenomena, of which a significant fraction were concluded to be "important" in prediction of SBWR transient and LOCA performance. TRACG contains models capable of simulation of each of the important phenomena, and each has been qualified by the successful predictions of at least one, and in most cases, several test data sets. The PIRT defines more than 900 specific data sets, from 42 different tests and test facilities, that make up the TRACG qualification database. Data from separate effects tests, component tests, systems and systems interaction tests, and operating plant experience have been predicted by TRACG in its validation.

Early in the SBWR program one piece of information was identified as needed for the SBWR for which there was no information in the database: that is, a heat transfer correlation for steam condensation in tubes in the presence of noncondensable gases. A test program has since been conducted to secure this information, reported to the NRC in Reference 19.

The Single Tube Condensation Test Program was conducted to investigate steam condensation inside tubes in the presence of noncondensables. The work was independently conducted at the University of California at Berkeley (UCB) and at the Massachusetts Institute of Technology (MIT). The work was initiated in order to obtain a database and a correlation for heat transfer in similar conditions as would occur in the SBWR PCCS tubes during a DBA LOCA. Three researchers utilized three separate experimental configurations at UCB, while two researchers utilized one configuration at MIT. The researchers ran tests with pure steam, steam/air, and steam/helium mixtures with representative and bounding flow rates and noncondensable mass fractions. The experimenters found the system to be well behaved for all tests, with either of the noncondensibles, for forced flow conditions similar to the SBWR design. The results of the tests at UCB have become the basis for the condensation heat transfer correlation used in the TRACG computer code.

While all SBWR features are extrapolations from current and previous designs, two features (specifically, the Passive Containment Cooling System and the Gravity-Driven Cooling System) represent the two most challenging extrapolations. Therefore, it was decided, for these two cases, to obtain additional test data, which could be used to demonstrate the capabilities of TRACG to successfully predict SBWR performance over a range of conditions and scales. Blind (in some cases double blind) predictions of test facility response use only the internal correlations of TRACG. No "tuning" of the TRACG inputs is to be performed, and no modifications to the coding are anticipated as a result of these tests.

For the case of the PCCS, it is planned to predict steady-state heat exchanger performance in full-vertical-scale 3-tube (GIRAFFE), 20-tube (PANDA), and prototypical 496-tube (PANTHERS) configurations, over the range of SBWR expected steam and noncondensable conditions (Appendix A). This process addresses scale and geometry differences between the basic phenomena tests performed in single tubes, and larger scales including prototype conditions. Transient performance is similarly investigated at two different scales in both GIRAFFE and PANDA.

TRACG GDCS performance predictions were performed against the GIST test series. Pre-test predictions have also been performed for the PANTHERS and PANDA steady-state tests.

1.2.2.1 Major SBWR-Unique Test Programs

As noted previously, the majority of data supporting the SBWR design came from the design and operating experience of the previous BWR product lines. SBWR-unique certification and confirmation tests are briefly described below. They will be discussed in detail in Appendix A to this report.

1.2.2.1.1 GIST

GIST is an experimental program conducted by GE to demonstrate the Gravity-Driven Cooling System (GDCS) concept and to collect GDCS flow rate data to be used to qualify the TRACG computer code for SBWR applications. Simulations were conducted of DBA LOCAs representing main steamline break, bottom drain line break, GDCS line break, and a non-LOCA loss of inventory. Test data have been used in the qualification of TRACG to SBWR and documented in Reference 42. Tests were completed in 1988 and documented by GE in 1989. GIST data has been used for validation of certain features of TRACG.

1.2.2.1.2 GIRAFFE

GIRAFFE is an experimental program conducted by the Toshiba Corporation to investigate thermal-hydraulic aspects of the SBWR Passive Containment Cooling System (PCCS). Fundamental steady-state tests on condensation phenomena in the PCC tubes were conducted. Simulations were run of DBA LOCAs; specifically, the main steamline break. Tests have been completed and results have been documented in Reference 43. GIRAFFE data will be used to substantiate PANDA and PANTHERS data at a different scale and to support validation of certain features of TRACG. Also, two additional series of tests will be conducted in the GIRAFFE facility: the first will demonstrate the operation of the PCCS in the presence of lighter-than-steam noncondesible gas; the second will provide additional information regarding potential system interaction effects in the late blowdown/early GDCS period.

1.2.2.1.3 PANDA

PANDA is an experimental program to be run by the Paul Scherrer Institut in Switzerland. PANDA is a full-vertical-scale 1/25 volume scale model of the SBWR system designed to model the thermal-hydraulic performance and post-LOCA decay heat removal of the PCCS. Both steady-state and transient performance simulations are planned. Testing at the same thermal-hydraulic conditions as previously tested in GIRAFFE and PANTHERS will be performed, so that scale-specific effects may be quantified. Blind pre-test analyses using TRACG will be submitted to the NRC prior to start of the testing. PANDA data will be used directly for validation of certain features of TRACG.

1.2.2.1.4 PANTHERS

PANTHERS is an experimental program to be performed by SIET in Italy, with the dual purpose of providing data for TRACG qualification and demonstration testing of the prototype PCCS and IC heat exchangers. Steam and noncondensibles will be supplied to prototype heat exchangers over the complete range of SBWR conditions to demonstrate the capability of the equipment to handle post-LOCA heat removal. Testing at the same thermal-hydraulic conditions as performed in GIRAFFE and PANDA is planned. Blind pre-test analyses of selected test conditions using TRACG have been submitted to the NRC prior to the start of testing [35]. PANTHERS data will be used directly for validation of certain features of TRACG.

In addition to thermal-hydraulic testing, an objective of PANTHERS is to investigate the structural adequacy of the heat exchangers. This objective is beyond the scope of this report.

1.2.2.1.5 Scaling of Tests

A discussion of scaling of the major SBWR tests is contained in Reference 32. That report contains a complete discussion of the features and behavior of the SBWR during challenging events. It includes the general (Top-Down approach) scaling considerations, the scaling of specific (Bottom-Up approach) phenomena, and the scaling approach for the specific tests discussed above. Appendix B supplements the scaling report with detailed quantitative analyses of the major SBWR test facilities.

Table 1.2-1 Evolution of the General Electric BWR

Product Line Number	Year of Introduction	Characteristic Plants/Features
BWR/1	1955	Dresden 1, Big Rock Point, Humboldt Bay, KRB, Dodewaard
		· Natural circulation (HB, D)
		· Internal steam separation
		· Isolation Condenser
		· Pressure suppression containment
BWR/2	1963	Oyster Creek
		· Large direct cycle
BWR/3/4	1965/1966	Dresden 2/Browns Ferry
		· Jet pump driven recirculation
		· Improved ECCS:spray and flood
		· Reactor Core Isolation Cooling System (replaced Isolation Condenser)(BWR/4)
BWR/5	1969	LaSalle
		· Improved ECCS systems
		· Valve recirculation flow control
BWR/6	1972	Grand Gulf
		· Improved jet pumps and steam separators
		· Improved ECCS performance
		· Gravity containment flooder
ABWR		· Internal recirculation pumps
		· Fine Motion Control Rod Drives
SBWR		· Gravity flooder, passive containment cooling
		· Return to Isolation Condenser
		· Return to natural circulation

Table 1.2-2 SBWR Features and Related Experience

SBWR Feature	Plants	Testing
IC	Dodewaard, Dresden 1,2,3, Big Rock Pt., Tarapur 1,2, Nine Mile Pt. 1, Oyster Creek, Millstone 1, Tsuruga, Nuclenor, Fukushima 1	Operating Plants
Natural Circulation	Dodewaard, Humboldt Bay	Operating Plants
Squib Valves	BWR/1-6 and ABWR SLC Injection Valves	Operating Plants IEEE 323 Qualification Testing
Gravity Flooder	BWR/6 Upper Pool Dump System, Suppression Pool Flooder System	Operating Plants Preoperational Testing
Internal Steam Separators	BWR/1-6 and ABWR	Operating Plants
Chimney (Core to Steam Separators)	Dodewaard, Humboldt Bay	Operating Plants
FMCRDs	ABWR	ABWR Test/Development Program (Demonstration at LaSalle Plant)
Automatic Depressurization Valves (MSIVs)	All BWRs	Operating Plants
Pressure Suppression	BWR/1-6 and ABWR	Mk I, Mk II, Mk III and ABWR Tests
Horizontal Vents	BWR/6 and ABWR	Mk III Testing ABWR Testing
Quenchers	BWR/2-6 and ABWR	Mk I/II/III Testing Operating Plants
PCC (Dual Function Heat Exchangers)	BWR/6, RHR HX Steam Condensing Mode	Operating Plants, PANDA, GIRAFFE, PANTHERS

Table 1.2-[illegible] SBWR and ABWR Analysis Methods

Analysis Type	Analysis Method	
	ABWR	***SBWR***
Steady-State	ISCOR/RODAN	ISCOR/TRACG
Transients		
· Pressurization	ODYN/TASC	TRACG
· Loss of Feedwater Heating	PANACEA	PANACEA
· Other	REDY/TASC	TRACG
ATWS	REDY/TASC	TRACG
Stability	FABLE/REDY	FABLE/TRACG
LOCA/ECCS	SAFER	TRACG
LOCA/Containment		
· Pressure/temperature response	M3CPT/SUPERHEX	TRACG
· Loads	Approved Methodology	Approved Methodology

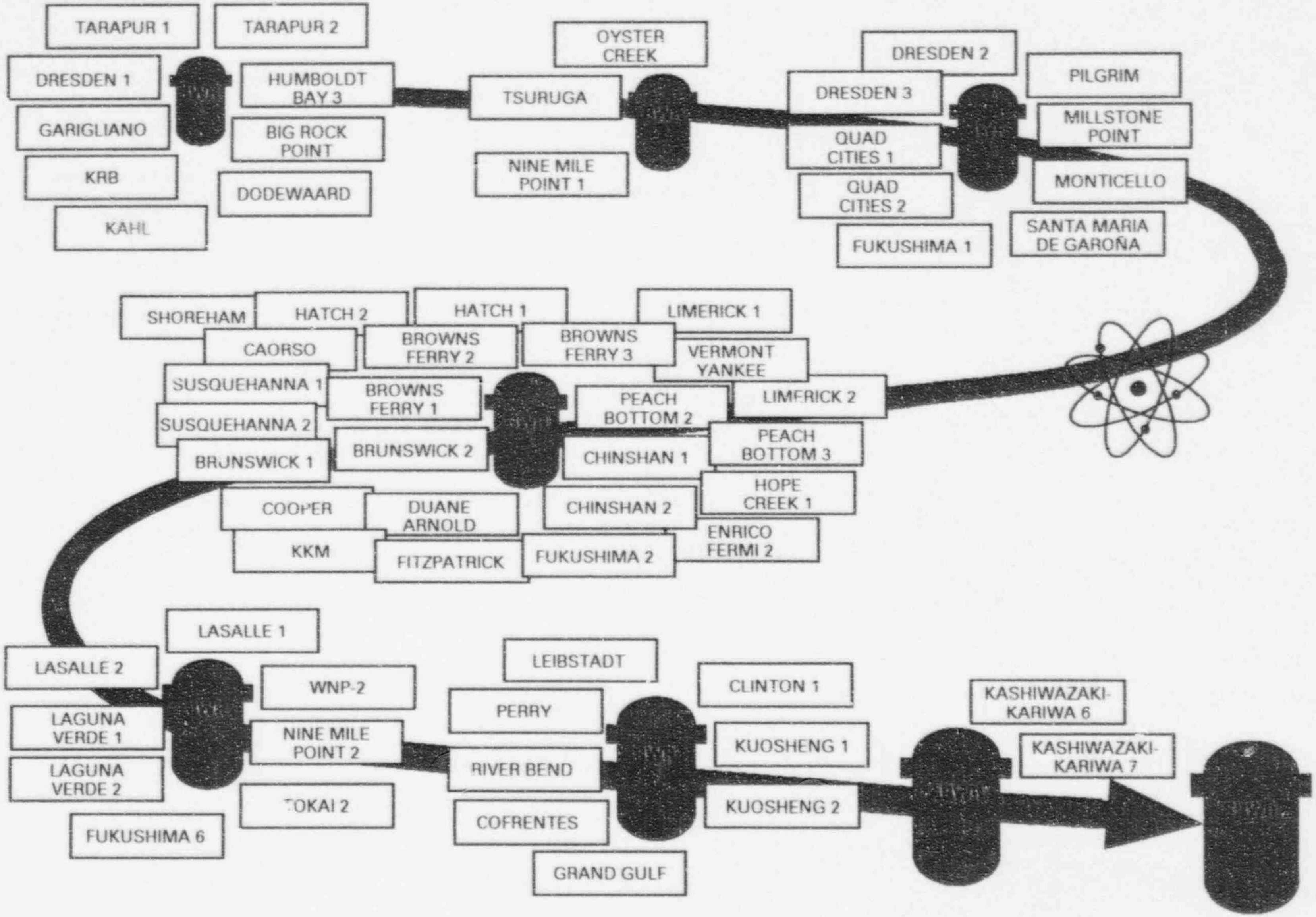


Figure 1.2-1 Evolution of the BWR

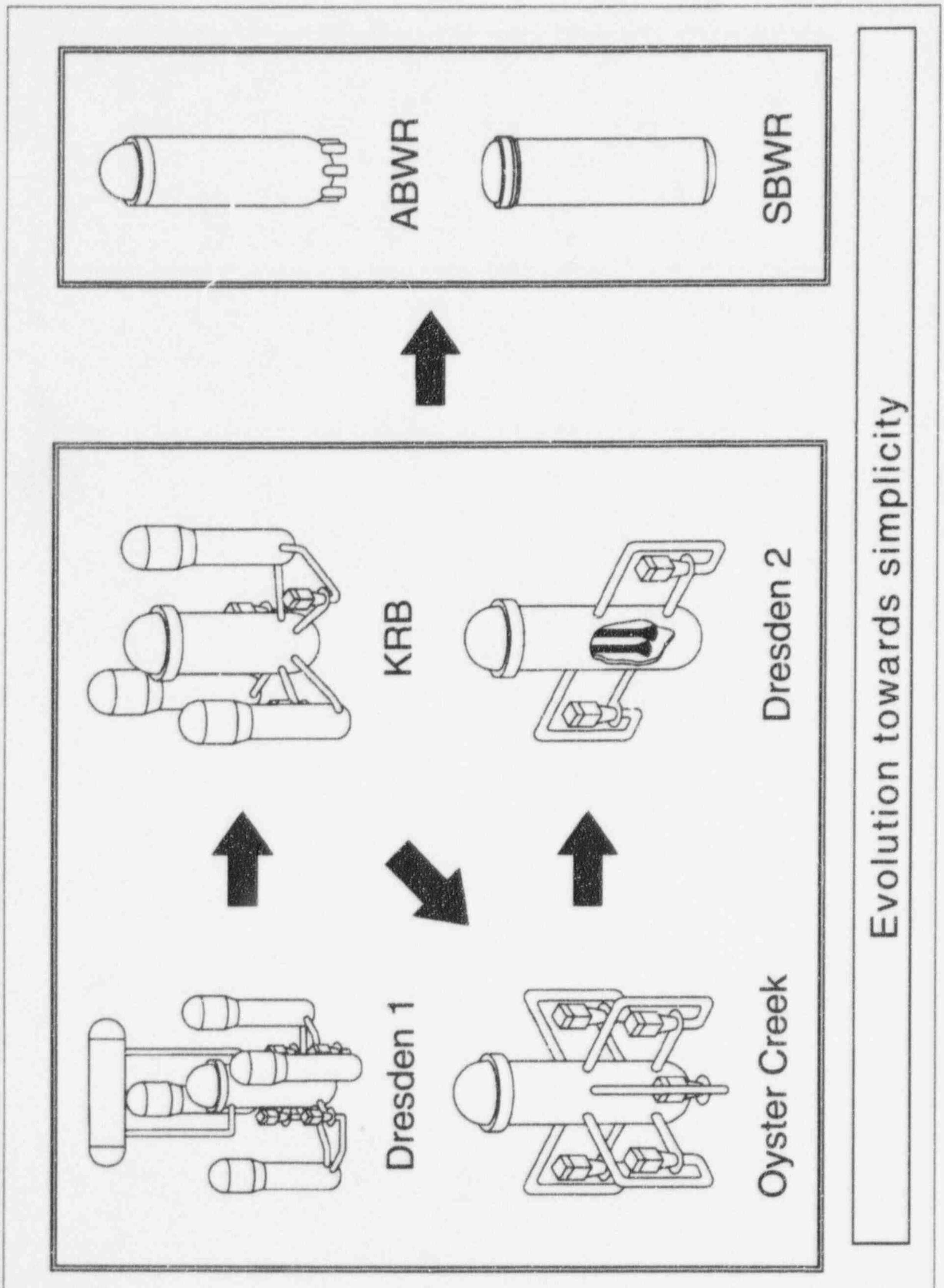
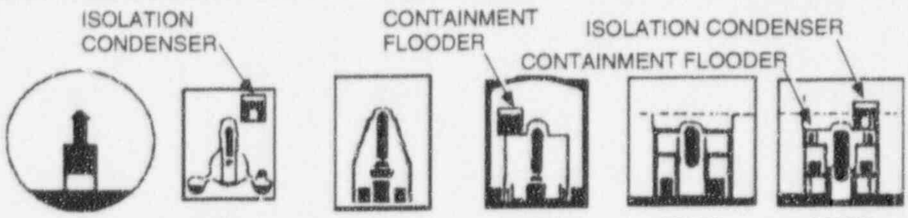


Figure 1.2-2 Evolution of the BWR Reactor



	DRY	MARK I	MARK II	MARK III	ABWR	SBWR
PRESSURE SUPPRESSION	NO	YES	YES	YES	YES	YES
NUMBER OF BARRIERS						
CONTAINMENT	1	2	2	3	2	2
FISSION	2	4	4	4	4	4
VOLUME (million ft^3)	2.5	0.4	0.5	1.6	0.5	0.3
HEAT CAPACITY (BTU $\times 10^9$)	0.3	1.7	1.3	1.3	1.3	1.3
DESIGN PRESSURE (psig)	50	62	45	15	45	55
LOCA PRESSURE (psig)	50	44	42	9	39	42

Figure 1.2-3 Comparison of BWR Containments

1.3 Strategy for Determination of Test and Analysis Needs

The process of defining test and analysis needs for analysis of SBWR transient and accident performance is based on developing a thorough understanding of the key phenomena to be simulated and modeled. Once such a list of phenomena and interactions between systems is compiled, the test and analysis plans can be checked against it to determine their sufficiency. In this study, a dual approach was used to arrive at a comprehensive list of controlling phenomena. Figure 1.3-1 shows the overall strategy. The Top-Down process starts with the calculated scenarios for the classes cf transients and accidents to be studied. The scenario is divided into different phases based on the key events in the evolution of the transient. For example, the LOCA/containment scenario can be divided into (1) the ***Blowdown phase***, where the reactor vessel depressurizes, enabling the Gravity-Driven Cooling System (GDCS) to start injecting water into the reactor vessel; (2) the ***GDCS phase*** during which the GDCS tanks drain into the reactor pressure vessel; and (3) the ***long-term cooling phase***, after the GDCS tanks have drained and the Passive Containment Cooling System (PCCS) removes decay heat and recycles condensed steam to the reactor vessel. For each phase of the transient, phenomena that might be important were listed and ranked to produce Phenomena Identification and Ranking Tables (PIRT). These tables were developed for each region of the reactor vessel and containment. This Top-Down process and the results are described in Section 2.

In the Bottom-Up process, unique SBWR design features were listed. Phenomena and issues related to these features that might influence SBWR operation and transient behavior were then compiled. This list was then reviewed and ranked by an independent team of experts. The resulting table of important phenomena and interactions is thus developed by an approach that is different from that used for the PIRT. Of course, both approaches require familiarity with SBWR transients and phenomena. This Bottom-Up process is described in Section 3.

The information developed through both approaches was combined into a comprehensive tabulation of SBWR phenomena. Because the Bottom-Up approach focused on SBWR-unique features, the PIRT contains 'generic' SBWR phenomena (common to all BWRs) that were not picked up by the SBWR-unique issues. On the other hand, because the Bottom-Up approach starts with specific SBWR components and systems, it was more suitable to identify interactions between components and the various SBWR systems. The composite table can be found in Section 4.1.

All the phenomena and interactions identified as important were evaluated. A Qualification Database sheet was prepared for each phenomenon, issue or interaction, showing the expected range of SBWR parameters, the range of test data available and an analysis of the adequacy of the database. This led to the identification of needs for additional test data or for TRACG qualification, which were factored into the test plan. The component and system interactions were also treated in the same manner. Numerous SBWR scenarios were analyzed to screen interactions that merited further study or experimental validation. This set was then compared with available integral system data that would capture these interactions. The test plan was amended to incorporate identified gaps in the database. The results of the analytical studies are summarized in Section 4.2. Further details on the calculations are contained in Appendix C.

The iterative evaluation process discussed above results in the TRACG Qualification Matrix (Section 5). The Qualification Matrix is a rearrangement of the Test Matrix showing how the identified phenomena are covered by specific tests. The Qualification Matrix has been divided into four categories: Separate Effects Data, Component Data, Integral System Data, and BWR Operating Plant Data.

The Test and Analysis Plan is discussed in Appendix A. It includes a brief description of each major SBWR test facility, and the test matrix, which contains the test conditions and the purpose and projected use for each category of tests. Planned analyses with TRACG for pre- and post-test calculations are identified. Detailed scaling studies were performed on the GIST, GIRAFFE, PANDA and PANTHERS facilities. The results show that the facilities are properly scaled to yield data for certification. Results of the scaling studies have been summarized in Appendix B.

Section 6 shows how the data will be used for TRACG development and validation. Separate effects and component data are used mainly for model development. Because interactions among components are present during the overall system response of integral test facilities, these data validate the overall performance of the TRACG Code for prediction of complex system response characteristics. Integral system tests provide confirmation of the validity of the models. The feedback from these tests may also be used to improve nodalization in the TRACG representation of the test facility and, possibly, the SBWR.

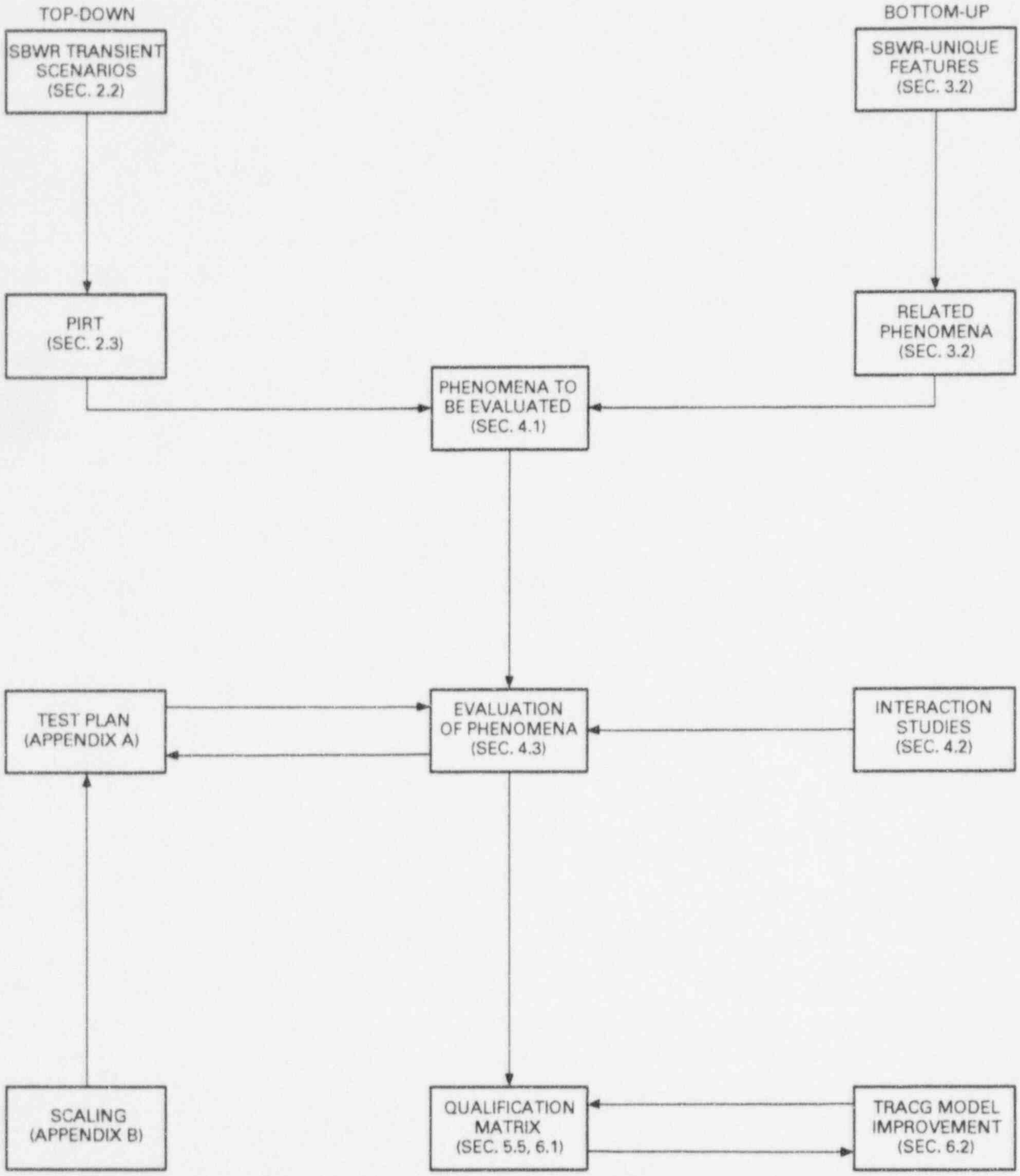


Figure 1.3-1 Strategy for Determination of Test Needs

1.4 Overall Test and Analysis Plan

This section shows the relationships between the various testing, qualification, licensing and design activities. In this study, the overall TRACG qualification needs are determined and additional SBWR related testing is defined as shown on Figure 1.4-1. As mentioned in the previous section, the primary output from the test and qualification activities is a final version of the TRACG computer program, which has been comprehensively validated for application to the SBWR. Figure 1.4-2 shows this process, which qualifies TRACG against large-scale component and integral system test data. A Licensing Topical Report describing TRACG Qualification against SBWR related test data will be prepared and submitted to the NRC for review and approval. Upon completion of the technology-related activities, the SSAR calculations in Sections 6 and 15 will be re-performed with the final version of the TRACG Code.

1.4.1 Relationship of TAPD Document to Overall TRACG Validation

TAPD describes the process for determining the necessary testing and analysis activities in support of SBWR technology and its application. The output from this document is a list of the required tests and analysis tasks. This report is supplemented by numerous other reports on test results, TRACG models, qualification and application methodology. The purpose of this section is to describe the various documents that are being submitted to the NRC for review, their relationships to one another, and their roles in providing the information needed for the validation and application of TRACG.

The CSAU road map, Figure 1.4-3 (from Reference 4), is a convenient means of describing how the necessary information is being provided. This road map identifies all the steps needed for validation and application of a computer code, starting from the selection of the application and the frozen code. The CSAU framework consists of three major elements comprising 14 steps. The first element relates to requirements and code capabilities. This is the process of defining the transient scenario to be analyzed (Step 1), selecting the nuclear power plant (Step 2), and development of the phenomena identification and ranking table (PIRT) (Step 3). A frozen version of the code is selected (Step 4) and the documentation is provided on the models in the code (Step 5). Comparison of the model capabilities with the phenomena to be modeled establishes the applicability of the code in Step 6. Element 2 is termed Assessment and Ranging of Parameters. The major steps in this element are to establish the assessment matrix (Step 7), perform assessment of the code against separate effects and integral effects tests to determine the appropriate nodalization to be used (Step 8), and to determine code biases and uncertainties (Step 9), as well as any bias and uncertainty due to the effect of scale (Step 10). The third element is comprised of sensitivity and uncertainty analyses. The effects of reactor input parameters and operating state are evaluated in Step 11 to determine code biases and uncertainties. Calculations (Step 12) are then performed to determine the sensitivity of key parameters to the various biases and uncertainties identified in Steps 9-11. These biases and uncertainties are combined in Step 13 to determine the total uncertainty for the transient under consideration (Step 14).

The TAPD addresses steps 1, 2, 3, 4, 6 and 7. PIRTs are developed for various transients, model capability is evaluated and the assessment matrix is established.

The TRACG models are described in Reference 1, TRACG Computer Code Model Description. This report was submitted to the NRC in February 1993 and is being revised to expand the description of the models and correlations. This report addresses Step 5 in Figure 1.4-3.

Reference 2, TRACG Computer Code Qualification, describes the developmental assessment of TRACG, as well as comparisons with separate effects tests, integral effects tests and BWR plant data. SBWR-specific facilities such as GIST and GIRAFFE are included in this list of comparisons. Several major SBWR related tests are currently underway. A supplementary report entitled "TRACG Computer Code Qualification for the SBWR", will be submitted after the tests are completed and analyzed. These two reports will address Steps 8, 9 and 10. In addition comparing the results of TRACG analyses with data, the nodalization to be used for reactor and containment analysis will be defined and model biases and uncertainties will be determined and included in the supplementary report.

Reference 7, Application of TRACG Model to SBWR Licensing Safety Analysis, is intended to address the remaining steps in the CSAU methodology (Steps 11 through 14). In the report previously submitted to the NRC, this process was completed for only operational transients. The report will be revised to incorporate the corresponding analysis for LOCA (ECCS and containment) application.

1.4.2 List of Reports to be Submitted to the NRC

The following is a list of Licensing Topical Reports planned to be submitted. The Tables of Contents for the Qualification LTR can be found in Appendix A, Attachment A1.

- TRACG Computer Code Model Description, NEDO-32176 and NEDE-32176P , Revision 1.
- TRACG Computer Code Qualification for SBWR, new. (Supplement to TRACG Qualification, NEDE-32177P)
- Application of TRACG to SBWR Licensing Safety Analysis, NEDO-32178 and NEDE-32178P, Revision 1
- SBWR Test Program, new.

Additional information will be provided through a number of supplemental reports. These consist of data reports and preliminary validation reports for each major test facility. A complete listing of these reports and their Tables of Contents are provided in Appendix A.

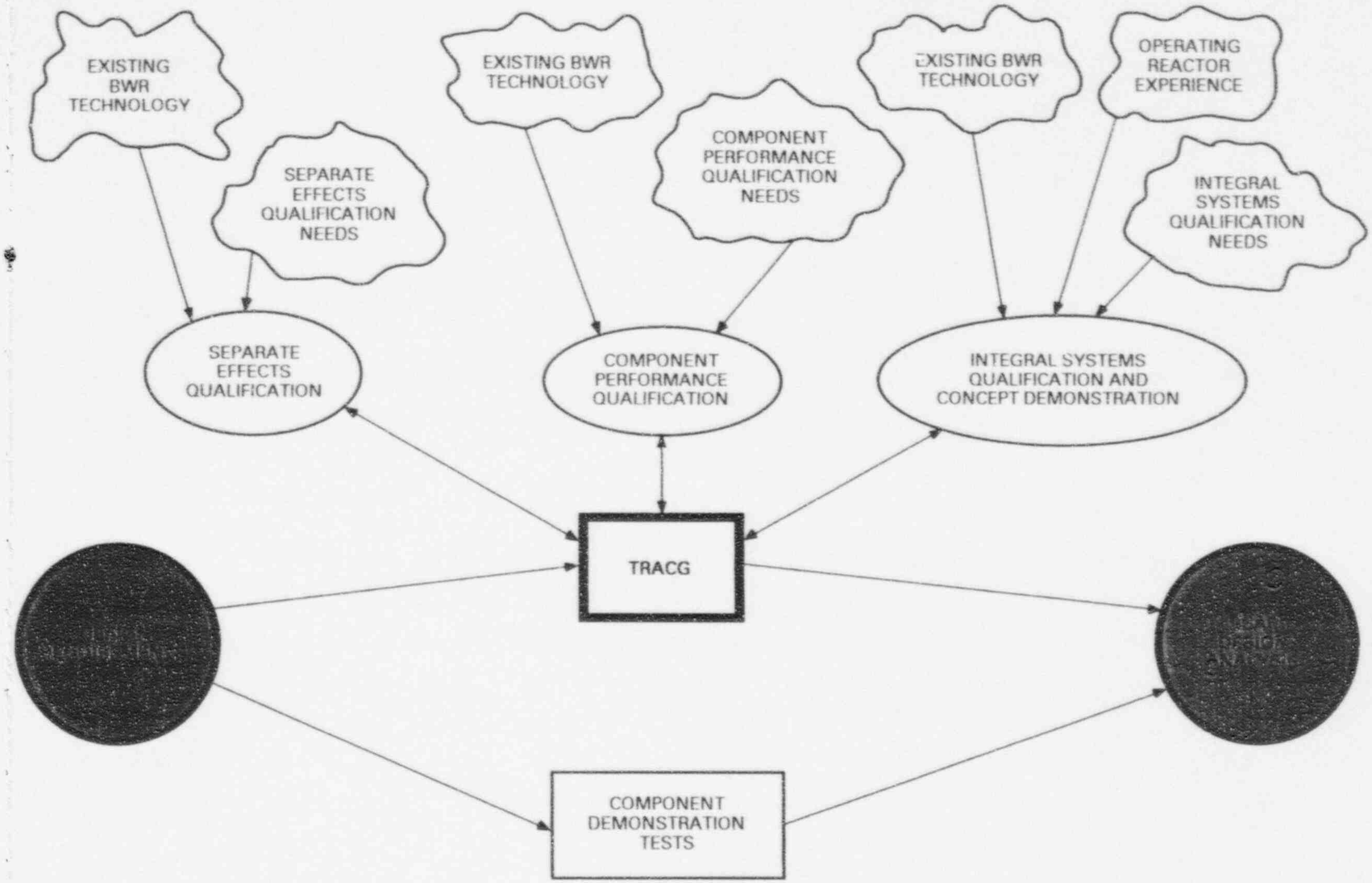


Figure 1.4-1 Technology Basis for SBWR Design

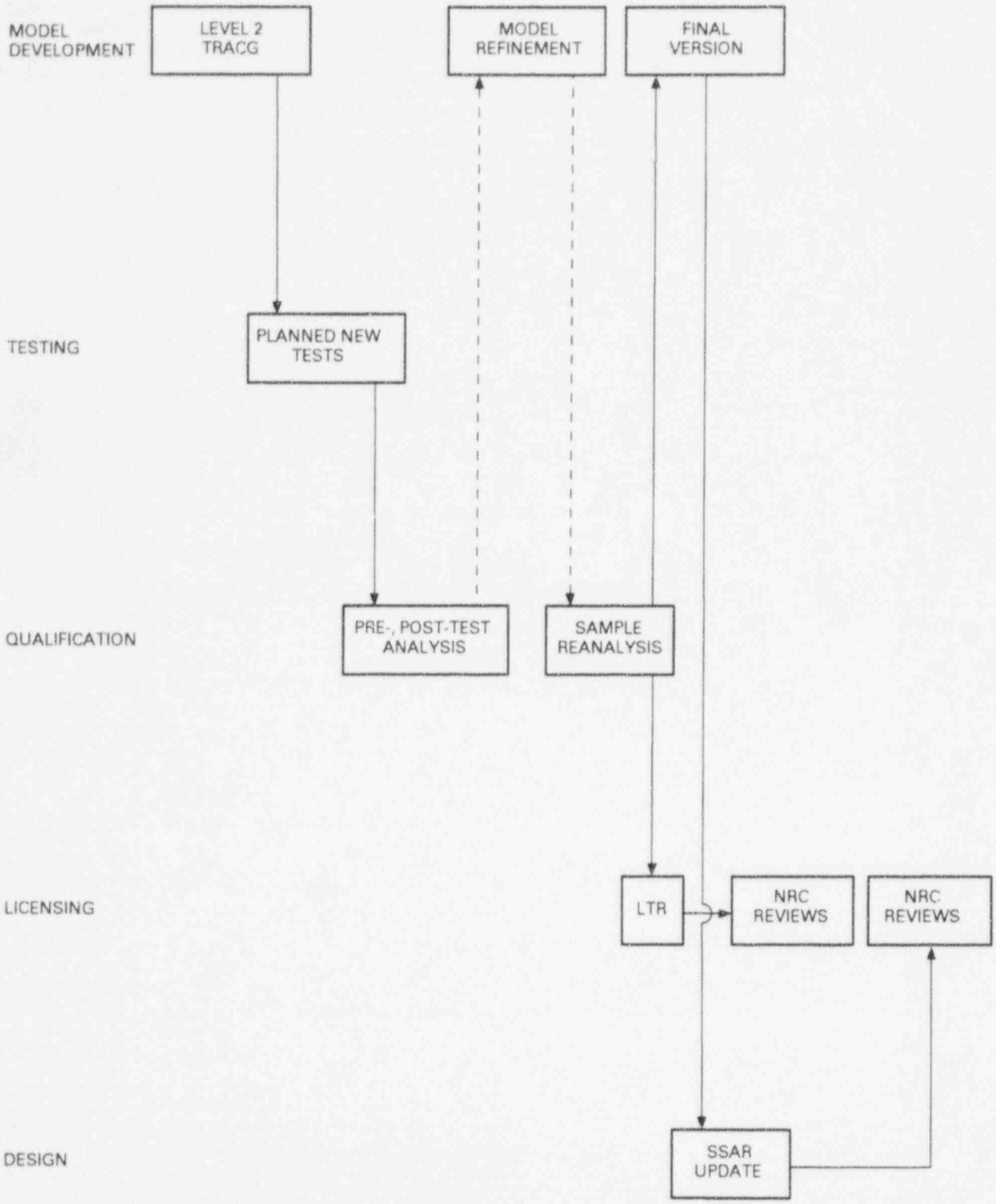


Figure 1.4-2 Overall Test and Analysis Plan

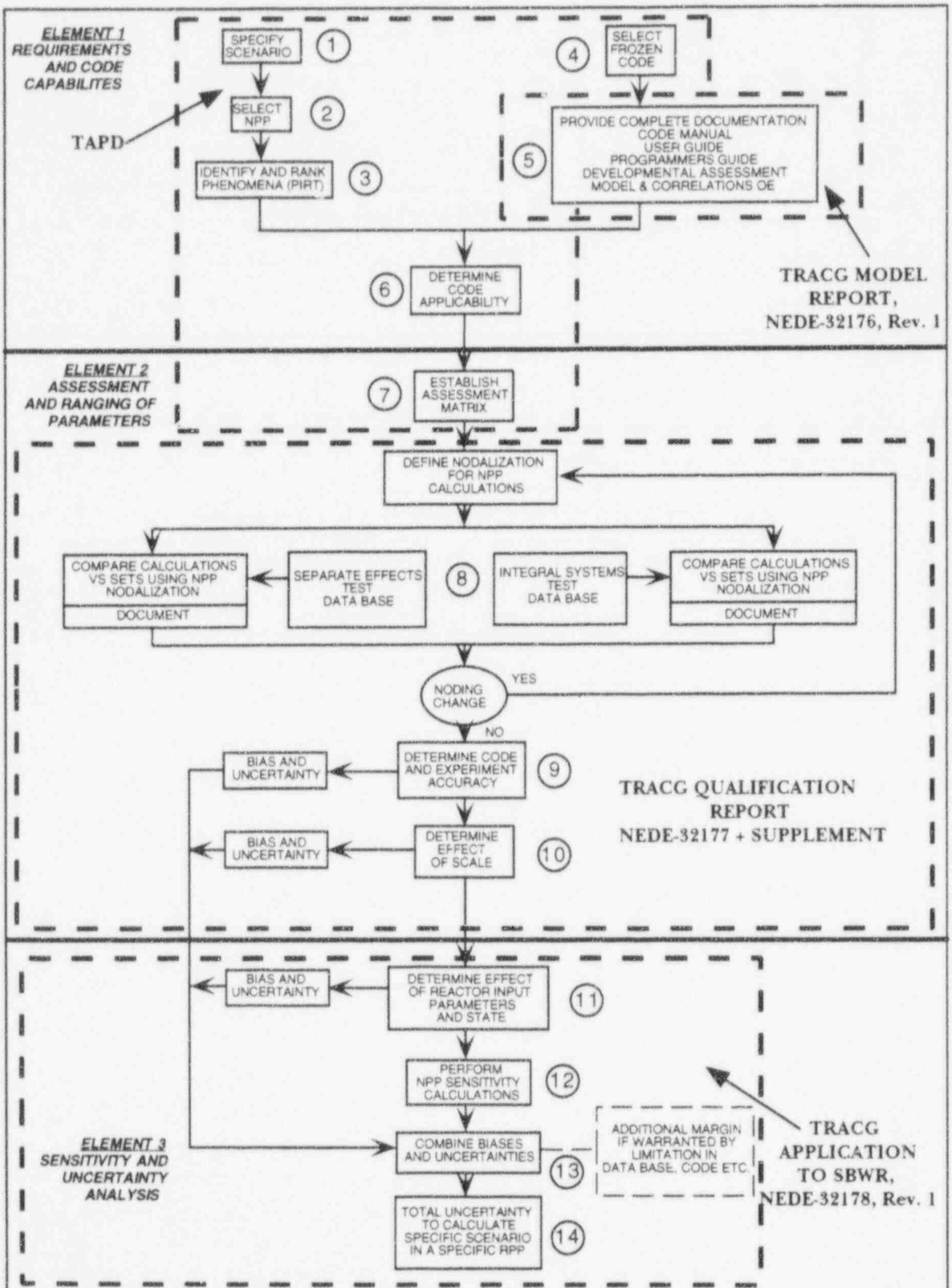


Figure 1.4-3. Road Map of SBWR TRACG Related Documentation

2.0 IDENTIFICATION OF IMPORTANT THERMAL-HYDRAULIC PHENOMENA: TOP-DOWN PROCESS

2.1 Introduction

As explained in Section 1.3 and illustrated in Figure 1.3-1, the process of defining test and analysis needs for analysis of SBWR transient and accident performance is based on developing a thorough understanding of the key phenomena to be simulated and modeled. This is done in this report in two ways: (1) a Top-Down process based on analyses and sensitivity studies, and (2) a Bottom-Up process based on examination of individual design features. The Top-Down process identifies phenomena and their importance based on how the overall system behaves; the Bottom-Up process, by component and subsystem requirements. This section discusses the Top-Down approach, leading to Phenomena Identification and Ranking Tables (PIRT). Chapter 3 discusses the Bottom-Up process. They are merged in Section 4.

The PIRT is a summary of analytical modeling needs for a physical system (in this case, the SBWR). The principal feature of the PIRT is an assessment of the "importance" of each modeling need by interdisciplinary teams of experts. The approach used in the SBWR follows the methodology of Boyack, et al [6]. TRACG calculations established the scenarios of various events (LOCA, anticipated transients, ATWS and stability). These are described in Section 2.2. The descriptions stress the phenomenological evolution of the transients. A detailed description of the sequence of events can be found in the SSAR [3]. (It is noted that, due to modeling and design changes since SSAR submittal, the event sequences have been updated somewhat from the SSAR versions.)

The analyses were then reviewed by interdisciplinary teams to identify each thermal-hydraulic phenomenon that plays a role in the analysis, and to rank all of them in terms of "importance"; that is, degree of influence on some figure of merit (e.g., reactor water level, containment pressure). The PIRT process is discussed in Section 2.3, where the PIRT tables are presented.

2.2 Analysis of Events

2.2.1 Loss-Of-Coolant Accident (LOCA)

Chapter 6 of the SSAR includes the entire matrix of calculations for postulated pipe rupture locations and single failures. For a complete PIRT evaluation, the entire spectrum of events must be covered, including analyses with less limiting conditions than the design-basis case with no auxiliary power. The approach followed in this study is to focus initially on the design basis cases, in terms of the equipment and systems available. This leads to the most severe consequences and the greatest challenges to the analytical models in modeling the phenomena. The next step was to examine the possible interactions with other systems that might be available, even though they are not classified as engineered safeguard features for the event. To facilitate understanding, a large break in the Gravity Driven Cooling System (GDCS) line has been chosen to illustrate the sequence of events during the LOCA. The sequence of events is similar for all the LOCA events, particularly after initiation of the GDCS flows, when the vessel and containment transients are closely coupled. While there are some differences in the assumptions made for analysis of the different breaks, these are not very important in determining the phenomenological progression of the LOCA or the importance of various parameters. The limiting LOCA from the perspective of margin to core uncovery is the GDCS line break; from the viewpoint of containment pressure, it is the large steamline break. A schematic of the SBWR's passive safety systems is shown in Figure 2.2-1.

The overall LOCA sequence can be divided into three periods: blowdown period, GDCS period and the long-term cooling PCCS period. These periods are shown in Figure 2.2-2. The ***blowdown period*** is characterized by a rapid depressurization of the vessel through the break, safety relief valves (SRVs) and depressurization valves (DPVs). The steam blowdown from the break and DPVs pressurizes the drywell, clearing the main containment vents and the PCCS vents. First, noncondensible gas and then steam flows through the vents and into the suppression pool. The steam is condensed in the pool and the noncondensible gas collects in the wetwell air space above the pool. At about 500 seconds, the pressure difference between the vessel and the drywell is small enough to enable flow from the GDCS pools to enter the vessel. This marks the beginning of the ***GDCS period***, during which the GDCS pools drain their inventory. Depending on the break, the pools are drained in between 2000 and 7000 seconds. The GDCS flow fills the vessel to the level of the break, after which the excess GDCS flow spills over into the drywell. The GDCS period is characterized by condensation of steam in the vessel and drywell, depressurization of the vessel and drywell and possible openings of the vacuum breakers which returns noncondensible gas from the wetwell airspace to the drywell. The decay heat eventually overcomes the subcooling in the GDCS water added to the vessel and boiloff resumes. The drywell pressure rises until flow is reestablished through the PCCS. This marks the beginning of the ***long-term PCCS cooling period***. During this period, the noncondensible gas that entered the drywell through the vacuum breakers is recycled back into the wetwell. Condensation of the boiloff steam in the PCCS is recycled back into the vessel through the GDCS pool. The most important part of the LOCA transient for vessel response is the blowdown period and the early part of the GDCS period when the vessel is reflooded and level restored. For some breaks, the equalization line from the suppression pool to the reactor vessel may open during the long-term cooling period to provide the vessel an additional source of makeup water.

2.2.1.1 Primary System Response for the GDCS Line Break

The GDCS line break scenario is a double ended guillotine break of a GDCS drain line. There are three GDCS pools in the SBWR containment, each with its own drain line from the pool to the vessel. Each drain divides into two branches before entering into the pressure vessel. Each branch has a check valve followed by a squib operated injection valve and finally a nozzle in the vessel wall to control the blowdown flow in case of a break. The check valve prevents backflow from the vessel to the pool. The GDCS break is assumed to occur in one branch, between the squib operated valve and the nozzle entering the vessel. Additional assumptions for the LOCA analysis include a simultaneous loss of auxiliary power and no credit for the on-site diesel generators. The only AC power assumed available is that from battery powered inverters.

- ***Blowdown Period*** — At break initiation, the assumed simultaneous loss of power trips the generator, causing the turbine bypass valves to open and the reactor to scram. The bypass valves close after 6 seconds. No credit is taken for this scram or the heat sink provided by the bypass. The power loss also causes a feedwater coastdown. Drywell cooling is lost and the control rod drive (CRD) pumps trip. The blowdown flow quickly increases the drywell pressure to the scram setpoint, although no credit is taken for this safety function.

  High drywell pressure isolates several other functions, including the Containment Atmosphere Control System (CACS) purge and vent, Fuel and Auxiliary Pool Cooling System (FAPCS), high and low conductivity sumps, fission product sampling, and reactor building Heating, Ventilating and Air Conditioning (HVAC) exhaust.

  Loss of feedwater and flow out the break cause the vessel water level to drop past the Level 3 (L3) scram setpoint. This setpoint is assumed to scram the reactor. The scram will temporarily increase the rate of level drop and the Level 2 (L2) trip will quickly follow the L3 trip. This trip will isolate the steamlines and open the isolation condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. After L2, the rate of level decrease will slow and, without external makeup, the Level 1 (L1) trip will be reached, but not for several minutes. During this delay, the IC, if available, would be removing energy and reducing pressure and break flow. After a 10-second delay to confirm the L1 condition, the Automatic Depressurization System (ADS) logic will start a timed sequential opening of depressurization and injection valves. Four SRVs (two on each steamline) open first. The remaining four SRVs open 10 seconds later to stagger SRV line clearing loads in the suppression pool and minimize vessel level swell. Similarly, opening of the depressurization valves (DPVs) is delayed 45 seconds. Two DPVs on the main steam lines open first, followed in 45 seconds by two additional DPVs. The remaining two DPVs open after an additional 45 seconds. Ten seconds after the last DPV opens, the six GDCS injection valves are opened. When the GDCS injection valves first open, the hydrostatic head from the pool is not sufficient to open the check valves and GDCS flow does not begin immediately. When the GDCS check valves do open, the cold GDCS water further depressurizes the vessel. Blowdown through the break and the SRVs and DPVs causes a level swell in the vessel, which collapses at the end of the blowdown period, with the GDCS injection.

- ***GDCS Period*** — The GDCS flow begins refilling the vessel and the downcomer level rises. When the level reaches the break, the GDCS flow spills back into the drywell. For the GDCS break, the flow of GDCS water is sufficient to raise the downcomer level above the break, until the pools empty, then the level drains back to the break level. Inside the core shroud, the level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. Figure 2.2-3 shows the chimney level during the first 25 minutes of the transient. The level swell during the initial blowdown and opening of the SRVs and DPVs is not shown in the figure (note the level drop and then rise during the GDCS period as the vessel is refilled).

2.2.1.2 Containment Response for the GDCS Line Break

Containment response calculations assume loss of all AC power except that available from battery powered inverters, reactor power at 102% of rated power and no credit for IC operation. The single failure used is the failure to open a check valve in one of the GDCS pool drain lines. Initial conditions are containment normal operating pressure and temperature, with the suppression pool at its maximum allowable operating temperature.

- ***Blowdown Period*** — The blowdown for the GDCS line break occurs from the vessel side of the broken line. Simultaneously, the pool side of the broken line drains the inventory of the one affected GDCS pool into the containment. The check valve keeps the vessel from blowing down through the unbroken branch of the GDCS line. As noted earlier, the break flow is initially a liquid blowdown, and after the downcomer water level falls below the GDCS line elevation, the break becomes a vapor blowdown. The ADS, activated by the downcomer level, opens the SRVs and the DPVs. The flashing liquid (and later, steam) entering the drywell increases its pressure, opening the main containment vents and sweeping most of the drywell noncondensible gas through the main vents, the suppression pool and into the wetwell airspace. The steam flow through the vents is condensed in the suppression pool. During the blowdown phase of the transient, the majority of the blowdown energy is transferred into the suppression pool through the main vents. Within the pool, temperature stratification occurs, with the blowdown energy being absorbed primarily in the region above the open vents. The increase in drywell pressure establishes flow through the PCCS, which also absorbs part of the blowdown energy. For the GDCS break, this period of the accident lasts less than 10 minutes. The peak containment pressure in the short term is primarily set by the compression of the noncondensibles initially in the drywell into the wetwell vapor space. The controlling parameters are the ratio of the drywell to wetwell vapor volumes, and the temperature at the top of the suppression pool, which sets the steam partial pressure.

- ***GDCS Period*** — Once the vessel pressure drops below the setpoint of the check valves in the two unbroken GDCS lines, the GDCS pools begin to empty their inventory into the vessel. The subcooled GDCS water quenches the core voids, stopping the steam flow from the vessel. The GDCS flow refills the vessel to the level of the break and then spills over into the drywell. Spillover from the break into the drywell begins at about 20 minutes into the accident and continues throughout the GDCS period of the accident. Once the GDCS flow begins, the drywell pressure peaks and begins to decrease. The decrease in drywell pressure stops the steam flow through the PCCS and main vents. The drop in drywell pressure is sufficient to open the vacuum breakers between the drywell and the wetwell airspace several times. Once the GDCS flow begins to spill from the vessel into the drywell, the drywell pressure drops further and additional vacuum breaker openings occur. Some of the noncondensible gas in the wetwell airspace is returned to the drywell through the vacuum breakers. The GDCS period of the transient continues until the GDCS pools empty and the decay heat is able to overcome the subcooling of the GDCS inventory in the vessel. Then, the drywell pressure rises and flow is re-established through the PCCS. The PCCS heat removal capacity, even while recycling noncondensible gas back to the wetwell, is sufficient to handle the steam generated by decay heat, and the main vents are not reopened. Any uncondensed steam condenses and deposits its latent heat in the portion of the suppression pool above the outlet of the PCCS vent. This period of the accident is expected to last approximately 3 hours for the GDCS line break.

- ***Long-Term PCCS Period*** — After the drywell pressure transient initiated by the GDCS flow is over, the drywell pressure settles out, slightly above the wetwell airspace pressure. A drywell-to-wetwell pressure difference is established which is sufficient to open the PCCS vent and drive the steam generated by decay heat through the PCCS. The drywell pressure and temperature during the first 12 hours of the GDCS line break transient are shown in Figure 2.2-4. The drywell pressure rises rapidly during the blowdown period, decreases at GDCS initiation, drops as the GDCS spills into the drywell and finally levels off as boiloff resumes. The temperature shown is for a node high in the drywell. At this location, the temperature rises during blowdown, then actually superheats during the GDCS period, but levels off as flow to the PCCS resumes. In lower regions of the drywell, affected by GDCS spill, the temperature may drop during the GDCS period. Figure 2.2-5 shows the PCCS power during the first 12 hours of the transient. Also shown is the decay heat. During the blowdown period, the PCCS picks up part of the energy released during the blowdown, most of which is deposited in the suppression pool. During the GDCS period, steam flow to the PCCS stops and the PCCS power drops to zero. As soon as the decay heat can overcome the GDCS subcooling, boiloff and steam flow to the PCCS resumes and by 12 hours, the PCCS power increases back to nearly equal to the decay heat power.

By way of comparison, the drywell pressure at the beginning of the long-term period for the GDCS line break is below the drywell pressure for the large steam line break. During the 72 hours which defines the long-term cooling period, the drywell pressure remains below the large steam line break pressure. As with other breaks, the drywell pressure established at the end of the GDCS period defines the containment behavior during the long-term cooling period.

For this particular break, depending on which GDCS line is broken, the vessel level may slowly drop during the long-term cooling period because part of the inventory that is boiled off and condensed in the PCCS may be returned to the GDCS pool with the break. This part of the PCCS flow will drain into the lower drywell instead of returning to the vessel. To avoid uncovering the core, an equalization line between the vessel and suppression pool is designed to open before the vessel water level can drop below one meter above the top of the core. This ensures sufficient liquid inventory to keep the core covered, even if the boiloff continues. For some breaks, the level in the lower drywell may rise enough to reach the spillover holes in the main vents. Inventory added to the lower drywell past this point is returned to the suppression pool and back to the vessel through the equalization line. Analysis of the GDCS break indicates that for this break, the drywell level will not reach the spillover holes.

During this final period of the transient, drywell pressure will rise slowly. This results from a slow increase in the wetwell airspace pressure, due to the assumed leakage flow between the drywell and wetwell airspace and conduction across the wall separating the drywell and wetwell. This energy addition is partially offset by heat losses to the surroundings from the outside wetwell wall. Without the leakage, the containment pressure remains nearly constant during the long-term period of the transient.

2.2.1.3 GDCS Line Break Summary

Although the discussion of the GDCS line break has been described in two parts, the primary system and containment response are not independent, particularly after the blowdown period. The sequence of events occurring in the GDCS line break transient is summarized in Table 2.2-1. The events which produce actions are listed as symptoms and the actions resulting from the event are listed as actions. The timing of the symptoms is also shown.

For the GDCS break, the reactor core does not uncover, so there is no cladding heatup above saturation temperature of the coolant. In evaluating the "importance" of various phenomena in the PIRT process, the phenomena associated with cladding heatup (e.g., radiation heat transfer, metal-water reaction) are comparatively unimportant, while phenomena associated with reactor water level (e.g., decay heat, energy release from heat slabs) are comparatively important. For the containment, after the blowdown and release of energy to the suppression pool, the effectiveness of the PCCS controls the containment response, with no pumped decay-heat removal system available. In the long-term cooling period, the containment pressure and temperature increase slowly until the end of the 72-hour period, at which time credit for non-safety decay-heat removal systems is permitted. Thus, containment pressure and temperature become the primary figures of merit for the containment and the phenomena affecting them are important.

The LOCA scenario develops slowly for the SBWR. The accident detection system logic functions almost instantaneously, but thereafter, the time scales are measured in hours rather than seconds. The reactor water level (Figure 2.2-3) dips briefly about 10 minutes into the LOCA due to void collapse following GDCS injection. For the GDCS line break, the minimum water level occurs at about 7 hours after the break. This slow response, which is due to the large volume of water in the reactor vessel and GDCS pools, makes the LOCA a very slow moving event from the reactor systems and operator response standpoint. Similarly, containment response (Figure 2.2-4) is gradual, not reaching the design pressure even 72 hours after the break. This slow response permits well-considered, deliberate operator actions.

2.2.1.4 Main Steam Line Break

In this subsection, the important features of the transient resulting from a large break in the main steam line are described. The emphasis is on those features that are different from the GDCS line break scenario.

- ***Blowdown Period*** — At break initiation, the blowdown flow quickly increases the drywell pressure to the scram setpoint, and a control rod scram occurs. The high velocities in the steam line initiate closure of the Main Steam Line Isolation Valves (MSIVs) and the reactor isolates in 3 - 5 seconds. This trip also opens the Isolation Condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. High drywell pressure isolates several other systems, including the Containment Atmosphere Control System (CACS) purge and vent, Fuel and Auxiliary Pool Cooling System (FAPCS), high and low conductivity sumps, fission product sampling, and reactor building Heating, Ventilating and Air Conditioning (HVAC) exhaust.

Loss of feedwater and flow from the break cause the vessel water level tc drop. Without external makeup, the Level 1 (L1) trip will be reached in about 6 minutes. During this period, the IC, if available, would be removing energy and reducing pressure and break flow. After a 10-second delay to confirm the L1 condition, the Automatic Depressurization System (ADS) logic starts a timed sequential opening of depressurization and injection valves. Two SRVs on the unbroken steam line open first. The remaining two SRVs open 10 seconds later to stagger SRV line clearing loads in the suppression pool and to minimize vessel level swell. The sequence of opening of the DPVs and the GDCS injection valves is similar to that for the GDCS line break described earlier. However, because of the large steam break, the vessel depressurizes faster and GDCS injection begins earlier, at about 500 seconds versus 600 seconds for the GDCS line break. Blowdown through the break, the SRVs, and the DPVs causes a level swell in the vessel. The level decreases at the end of the blowdown period, when GDCS injection begins.

In the containment, the steam entering the drywell increases its pressure, opening the main containment vents and sweeping most of the drywell noncondensible gas through the main vents, through the suppression pool, and into the wetwell airspace. During the blowdown phase of the transient, the majority of the blowdown energy is transferred into the suppression pool by condensation of the steam flowing through the main vents. The increase in drywell pressure causes flow through the PCCS, which also absorbs part of the blowdown energy. The ADS, activated by the downcomer level, opens the SRVs and the DPVs and augments the steam flow to the suppression pool and drywell, respectively. This period of the accident lasts less than 10 minutes.

- ***GDCS Period*** — The GDCS flow begins refilling the vessel and the downcomer level rises. When the level reaches the elevation of the open DPVs, the GDCS flow spills back into the drywell. Inside the core shroud, the level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. The minimum water level in the chimney is of the order of 3-4 m above the top of the core; there is substantial margin to core heatup.

  Quenching of voids in the core by the GDCS flow reduces the steam outflow from the vessel to the drywell. Once the GDCS flow begins, the drywell pressure peaks and begins to decrease. The decrease in drywell pressure stops the steam flow through the PCCS and main vents. This pressure decrease may be sufficient to open the vacuum breakers between the drywell and the wetwell airspace. Once GDCS flow begins to spill from the vessel into the drywell, the drywell pressure drops further and additional vacuum breakers may open. If the vacuum breakers open, some of the noncondensible gas in the wetwell airspace will return to the drywell through the vacuum breakers. The GDCS period of the transient continues until the level in the GDCS pools equalizes with that in the reactor pressure vessel and the decay heat is able to overcome the subcooling of the GDCS inventory in the vessel. Then, the drywell pressure rises and flow is re-established through the PCCS. The PCCS heat removal capacity, even while recycling noncondensible gas back to the wetwell, is sufficient to transfer the steam generated by decay heat without reopening and the main vents. This period of the accident is expected to last for less than one hour.

- ***Long-Term PCCS Period*** — After the drywell pressure transient initiated by the GDCS flow is over, the drywell pressure settles out, slightly above the wetwell airspace pressure. The Main Steam Line break is the limiting break in terms of containment pressure and temperature. This part of the containment transient is similar to that for the GDCS line break. However, unlike the GDCS line break, the steam generated by the decay heat is condensed and all of it is returned to the vessel through the GDCS lines. Thus, there is no long term drop in the vessel water level due to boiloff. A larger amount of water inventory is retained inside the vessel and a smaller amount in the lower drywell.

2.2.1.5 Small Breaks

The thermal hydraulic phenomena which characterize the small breaks in the SBWR are very similar to those for the large steam line break. This is because once the downcomer level drops below the Level 1 set point, the reactor is automatically depressurized through the SRVs and DPVs. For small breaks (depending on the size and location), it may take several minutes before the reactor is scrammed on low water level (Level 3), and still longer before the ADS is actuated. For a steam line break having an area equivalent to 2% of the main steam line cross-sectional area, the reactor water level will boil off to reach Level 1 in about one hour. During this period, the break flow exceeds the condensing capacity of the PCCS and results in clearing the top row of horizontal vents. This results in energy addition to the portion of the suppression pool above the top vents, and increases the pool surface temperatures. The SBWR incorporates an ADS trip on high pool surface temperature to mitigate this effect.

2.2.1.6 Non-Design Basis LOCAs

The discussion to this point has focused on LOCA scenarios with design basis assumptions. With regards to system availability, the primary assumptions were to assume failure in an active system or component and loss of offsite power and diesel generators. The consequences of relaxing these assumptions towards a "best estimate scenario" are examined in this subsection.

Single Failures:

In the SBWR, the active component failures considered are the failure of a valve in the GDCS line to open and the failure of a DPV to open. Scenarios without failures have been analyzed. With no failures, design margins are increased. No new thermal-hydraulic phenomena or interactions are introduced because the differences relate simply to the number of GDCS lines available (quantity of GDCS flow) or the number of DPVs available for depressurization (amount of steam blowdown flow and rate of depressurization). While no new phenomena are introduced, these events do provide a wider range of parameters which is useful for code validation. Tests with both types of single failure and ones without any failure are included in the LOCA simulations performed in the GIST facility.

Isolation Condenser Operation:

For LOCA analysis, the IC is not treated as an engineered safety feature and no credit is taken in the safety analysis for its operation. The valve in the condensate return line will open in a realistic scenario. This increases the vessel liquid inventory before ADS and reduces the steam load on the containment. LOCA scenarios with the IC operational have been included in the consideration of important phenomena in Sections 3 and 4. These phenomena include the IC condensation efficiency, steam quenching in the reactor vessel downcomer, and interactions between the IC steam flow and the steam flow through the DPVs on the same nozzle.

Diesel Generators Available:

As shown in Table 2.2-2, additional systems become available when the diesel generators start up. Only the Control Rod Drive System in its high pressure injection mode is initiated automatically. This system injects water through the feedwater line into the downcomer. Scenarios with the CRD high pressure injection available are considered in Chapter 3 and Section 4.2. The Fuel and Auxiliary Pool Cooling System (FAPCS) will also be available to the operator with the diesels operational. FAPCS isolates automatically on high drywell pressure. The operator can override the isolation manually. The FAPCS has several modes of operation. It can be aligned to function initially in the Low Pressure Coolant Injection (LPCI) mode. When core cooling is established, the FAPCS can serve as a Suppression Pool cooling system. It can also be used for drywell and wetwell spray. Interactions between the FAPCS and the passive safety systems (GDCS/PCCS) are considered in Chapter 3 and analyzed in detail in Section 4.2.

Offsite Power Available:

Table 2.2-3 shows that the primary additional water makeup systems available with offsite power are the condensate and feedwater systems. Numerous auxiliary systems such as fuel pool cooling, drywell coolers, and drywell sump drain pumps would also be available. With feedwater and offsite power available, the accident becomes a relatively mild event. After scram on high drywell pressure, the feedwater maintains normal water level for an extended period of time even for large breaks. This allows the operator to initiate a controlled depressurization of the reactor. The water spilling out of the reactor collects in the lower drywell. For large breaks, the sump drain pumps will not be able to keep up with the break discharge. Eventually, water spills into the wetwell through the spillover holes in the pipes connected to the horizontal vents. The feedwater will be throttled back or turned off as the level rises in the wetwell.

2.2.2 Anticipated Transients

As with the LOCA, anticipated transients are discussed in the SSAR (Chapter 15) and results for specific events are not presented in this report. The PIRTs for anticipated transients were synthesized from consideration of the phenomena involved in various classes of events.

2.2.2.1 Fast Pressurization Events

These are the limiting pressurization events. Principal figures of merit on which "importance" is defined are critical power (MCPR) and reactor pressure.

- ***Turbine Trips*** — initiated by trip of turbine stop valves from full open to full closed. Analyzed with bypass valves functional, and with bypass failure.
- ***Generator Load Rejection*** — initiated by fast closure of turbine control valves from partially open position to full-closed. This event is analyzed with bypass valves functioning, and with bypass failure. The turbine control valves may be initially at the same position (full arc turbine admission) or at different positions (partial arc turbine admission).
- ***Loss of AC Power*** — Similar to load rejection; however, bypass valves are assumed to close after 6 seconds due to loss of power to condenser circulating water pumps.
- ***Main Steamline Isolation Valve (MSIV) Closure*** — In this case, the scram signal on valve position is further in advance of complete valve closure. This effectively mitigates the shorter line length to the vessel available as a compression volume.
- ***Loss of Condenser Vacuum*** — This event is similar to the Loss of AC Power and a Turbine Trip with Bypass. Because a turbine trip occurs at a higher vacuum setpoint than the bypass valve isolation, the bypass valves are available to mitigate the initial pressure increase.

2.2.2.2 Slow Pressurization Events

These are analyzed principally to ensure that they are bounded by the fast pressurization events. MCPR and reactor pressure determine "importance."

- ***Pressure Regulator Downscale Failure*** — Simultaneous closure of all turbine control valves in normal stroke mode. The triplicated fault tolerant control system prevents any single failure from causing this and makes its frequency below the anticipated abnormal occurrence category.
- ***Single Control Valve Closure*** — This event could be caused by a hydraulic failure in the valve or a failure of the valves rotor/actuator.

2.2.2.3 Decrease in Reactor Coolant Inventory

Loss of feedwater flow is characteristic of this category of transient. The IC maintains water level. Reactor water level is the principal figure of merit on which "importance" is defined.

2.2.2.4 Decrease in Moderator Temperature

These events challenge MCPR and stability, which are the figures of merit on which "importance" is defined:

- ***Loss of Feedwater Heating*** — initiated by isolation or bypass of a feedwater heater.
- ***Feedwater Controller Failure*** — hypothesizes an increase in feedwater flow to the maximum possible with all three feedpumps operating at maximum speed. Similar to turbine trip but with more severe power transient due to colder feedwater.

To determine the phenomena important in modeling anticipated transients, the sequence of events and system behavior for each class of events should be understood. To provide an example of this, the sequence of events for a fast pressurization transient is discussed below. For this class of transients, important phenomena are those affecting the MCPR and reactor pressure.

2.2.2.5 Generator Load Rejection Event Description

A fast pressurization event will occur due to the fast closure of the turbine control valves (TCVs), which can be initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. Closure of the turbine stop valves is initiated by the turbine protection system. The valves are required to close rapidly to prevent excessive overspeed of the turbine-generator rotor.

At the same time, the turbine stop or control valves are signaled to close, and the turbine bypass valves are signaled to open in the fast opening mode. The bypass valves are full open only slightly later than the turbine valves are closed, and can relieve more than one-third of rated steam flow to the condenser, greatly mitigating the transient. The bypass valves also use a triplicated digital controller. No single failure can cause all turbine bypass valves to fail to open on demand. The worst single failure can only cause one turbine bypass valve to fail to open on demand.

The closing time of the TCVs is short relative to the sonic transit time of the steamline, so their closure sets up a pressure wave in the steam lines. When the pressure wave reaches the vessel steam dome, the flow rate leaving the vessel effectively undergoes a step change. The area change entering the steam dome partially attenuates the pressure wave, propagating a weaker pressure disturbance down through the chimney and downcomer, increasing the vessel pressure, and reducing voids in the core. The void-reactivity feedback results in an increase in the neutron flux. A reflection of the pressure wave also travels back toward the turbine, producing an oscillation in flow and pressure in the steam lines.

Concurrent with closure of the turbine control valves, a scram condition is sensed by the reactor protection system. A turbine stop valve position less than approximately full open triggers a scram, as does the low hydraulic fluid pressure in the turbine control valve solenoids which start their fast closure mode. The SBWR digital multiplexed Safety System Logic Control (SSLC) will initiate a scram when any two turbine stop valves are sensed as closing, or any two turbine control valves are sensed as fast closing.

The core reactivity is decreased by the control blade insertion and increased by the decrease in core voids and increase in inlet flow. The net effect may be either an immediate shutdown of the reactor and decrease in neutron flux (in cases where there are control blades partially inserted in high worth areas of the core) or a short period of increased reactivity and neutron flux followed by shutdown (in the safety analysis case where there are no control blades initially inserted, and a slower bounding CRD scram insertion time is assumed.)

In the case where the neutron flux undergoes a transient increase, the energy deposition in the fuel pellet will increase clad heat flux. The minimum value of critical power ratio during this transient is found to occur in the upper part of the bundle.

Eventually, as the blades are fully inserted, the reactor is driven subcritical, power drops to decay heat levels, and clad temperature equilibrates near saturation temperature.

The vessel pressure increase is terminated by the bypass valve opening. The water level drops below the feedwater sparger and sprays subcooled water into the steam dome. This quenching of vapor also helps to terminate the pressure increase. If the bypass and feedwater systems are assumed to be unavailable, the duration of increased pressure would be long enough to initiate the isolation condenser.

In the ASME overpressure protection analysis, the Isolation Condenser is not considered, causing the pressure to slowly increase to the SRV opening pressure. The pressure increase is terminated immediately with SRV activation, and the maximum vessel pressure occurs at the vessel bottom. The overpressure protection case conservatively assumes the first scram signal to fail, and scram on neutron flux terminates the power increase in both turbine valve closure and the MSIV closure events.

The water level response in pressurization events is driven by the transfer of water from the downcomer to core and chimney caused by the collapse of voids in the core and chimney regions. The sensed water level decreases rapidly below the L3 low water scram setpoint. The feedwater system flow increases fast enough to prevent the L2 setpoint being reached in high frequency events (events where feedwater and bypass valves are available). The feedwater control system will demand maximum feedwater flow for approximately one minute, until normal level is restored. Without feedwater, the level drop will progress to L2, initiating the IC, isolating the MSIVs and transferring the CRD system to high pressure injection mode. The IC can independently maintain the water level near the L2 setpoint. CRD high pressure injection will cause level to slowly recover to above normal, and then automatically trip off.

2.2.3 Anticipated Transients Without Scram (ATWS)

The most limiting ATWS event in terms of reactor vessel pressure, heat flux, neutron flux, peak cladding temperature, suppression pool temperature and containment pressure is the inadvertent closure of all main steamline isolation valves with failure of rod insertion. This event is described in Section 15.8 of the SSAR. It is the only ATWS event considered in determining the phenomena needs for qualification of TRACG.

(A more detailed description of the MSIV closure ATWS will be provided in Revision C of this document, with emphasis on the important phenomena that characterize this transient.)

2.2.4 Stability

Because the SBWR core flow is driven by natural circulation, the most limiting stability condition is at the rated power/flow condition. This is unlike operating forced-circulation BWRs, and it simplifies the stability analysis for the SBWR.

For the SBWR, a stability criterion is used which is very conservative compared to operating plants (Figure 2.2-6). The core decay ratio is maintained less than 0.4 and the channel decay ratio less than 0.3.

The stability performance of the SBWR is evaluated at various conditions.

2.2.4.1 For Steady-State Operation

During *steady-state operation*, the highest power/flow ratio occurs at 104.2% power and 100% flow conditions. The decay ratio is well within the conservative design criteria (Figure 2.2-6). At reduced power level, the power/flow ratio is lower, so the decay ratios for both core and hot channel are lower than at the rated condition. This conclusion is supported by Dodewaard test data as shown in the figure. The decay ratios during normal operation at Dodewaard have been very low, with no indication of any incipient instability throughout its long operating history. In Figure 2.2-7, the power/flow map of SBWR normal operation is compared with the stability limit calculated in the Oak Ridge National Laboratory (ORNL) study. The results confirm that there is large margin for stability. This indicates that the SBWR is very stable under normal operation conditions.

2.2.4.2 For Anticipated Transients

Of the *anticipated transients*, the loss of 55.6°C (100°F) feedwater heating case gives the highest power/flow ratio. Loss of feedwater flow is another limiting event. However, the scram quickly mitigates the transient and the power conditions are reduced to hot shutdown. For both events, the decay ratios for core and hot channel meet the design criteria shown in Figure 2.2-6. In Figure 2.2-7, both of these transient events are seen to result in power/flow conditions that are well below the exclusion region.

2.2.4.3 For ATWS Conditions

During *ATWS* conditions, the persistent high reactor power poses the most challenge to the stability criteria. However, feedwater runback reduces the core power, and the SBWR's low power density also helps to alleviate the severity of the challenge to the stability criteria. Even though the reduced vessel water level effectively decreases the core flow rate and increases the power/flow ratio to a higher value than those for the steady state and anticipated transient conditions, the analysis of performance in the ATWS study indicates the reactor remains stable and no power oscillation is predicted. Following the feedwater runback, both flow and power decrease, resulting in a more favorable power/flow ratio. The injection of boron will eventually shut down the reactor and terminate the transient.

2.2.4.4 For Startup

During *startup*, there is a special concern that is not present at power. At very low flows, a periodic "geysering" flow oscillation can be postulated to occur caused by either of two mechanisms. First, condensation of core exit vapor in the subcooled chimney region and the top of the core might cause a reduced pressure in the channels and a resultant flow reversal in the core. Oscillations of this kind are unlikely given the SBWR startup procedures, which are similar to those of the Dodewaard reactor (Dodewaard has experienced no "geysering" oscillation in its 22 refuel cycles of operation). Second, vapor production in the lower-hydrostatic-head chimney region could cause a reduction of hydrostatic head and a resultant core flow increase. This, in turn, could cause voids to collapse in the chimney, leading to a reduction in flow. Oscillations of this second kind have also never been seen at Dodewaard. If they were to occur, they would be mild oscillations with little, if any, reactivity impact.

Table 2.2-1 GDCS Line Break Sequence of Events

Symptom	Action(s)	Time (hr)
Loss of offsite power	Instantaneous GDCS line break. Generator trips, bypass valves open and reactor scrams. Bypass valves close after 6 seconds. No credit for this scram or the bypass heat sink is taken in the SSAR Chapter 6 analysis	0.
	Feedwater coastdown (diesel generators fail to start)	
	Fuel pool cooling lost	
	DW coolers lost	
	CRD pumps trip	
High drywell pressure	Scram (no credit taken)	0.01 (Note 1)
	CACS (Cont. Atm. Control Sys) purge & vent isolates	
	FAPCS (Fuel and Aux. Pool Cooling Sys.) isolation	
	PCC condensation begins	
	PCC pool boiloff begins, HX tubes remain covered >72 hr	
	Isolate high and low conductivity sumps, fission product sampling, reactor building HVAC exhaust	
Low water level L3	Scram	0.01 (Note 1)
Low water level L2	IC drain valve opens (MSIV closure also initiates)	0.01 (Note 1)
	Isolate high and low conductivity sumps, fission product sampling, reactor building HVAC exhaust	
	DW coolers isolate	
Low water level L1	ADS/GDCS initiation. Timed sequential opening of: 4 SRVs/4 SRVs/2 DPVs/2 DPV's/2 DPV's/6 GDC injection valves	0.1
	DW coolers isolate	
	Same equipment which isolated on L2 receives redundant isolation signal.	
P < GDC pool head	Injection flow begins	0.2
Post LOCA radiolytic H_2 and O_2	PARs (Passive Autocatalytic Recombiners) function. (PARs are not simulated in fuel peak temperature and minimum water level calculations)	0.2 (Note 2)
P dw < P ww – 0.5 psi	Vacuum breakers open	0.3
GDCS pool empties	DW pressure stabilized	2.4
	DW-WW Δp initiates PCCS flow	
	PCCS condensate returns to GDCS pool, drains to vessel and DW	
Reactor water level falls to one meter above top of core	Vessel to S/P equalization line opens, keeps core covered	6.6
Liquid in DW reaches spillover holes in main vents	Inventory added to DW now returns to S/P (then to vessel)	9.3 (Note 3)
Design-basis leakage and sensible heat transfer from DW to WW causes gradual increase of DW pressure	Pressure rises slowly for 72 hours (defined as end of design basis)	to 72

NOTES To Table 2.2-1:

(1) Scram on high drywell pressure and level decrease to L2 occur within one minute of the line break.

(2) PARS will actuate as soon as they are exposed to radiolytic hydrogen, estimated to occur within a few minutes of the line break.

(3) Increase of DW level to the spillover holes only occurs if it is assumed that inward flow through the break cannot occur. Otherwise, the inventory spilled to the DW returns to the RPV through the break.

Table 2.2-2 LOCA Scenario with Diesel Generators Available - Additional Systems Functional

Symptom	Action(s)
Loss of normal AC	Diesel Generator starts
	FMCRD run-in backs up hydraulic scram
Low water level L2	CRD initiates in high pressure injection mode
Above actions are automatic, no operator action necessary. Actions below require operator intervention.	
Low water level L3	FAPCS LPCI mode, injection through FW system
High pool temperature	FAPCS Pool cooling mode, if adequate core cooling. Operator action required to over-ride system isolation.
P cont > 14.2 psig	FAPCS DW and WW spray
T dw > ADS qualification temperature	FAPCS drywell spray
Low water level < L1 per EPG	Firewater
Containment pressure high or T dw > Tech Spec LCO	DW Cooler
GDCS Pool level < NWL - 0.5m (2 of 3 pools)	Trip CRD pumps
2 days post LOCA	Attach PCC vent fan

Table 2.2-3 LOCA Scenario with Offsite Power & Diesel Generators Available -

Symptom	Action(s)
Low water level L3	FW and condensate injection
Pressure > normal setpoint	Turbine bypass valves

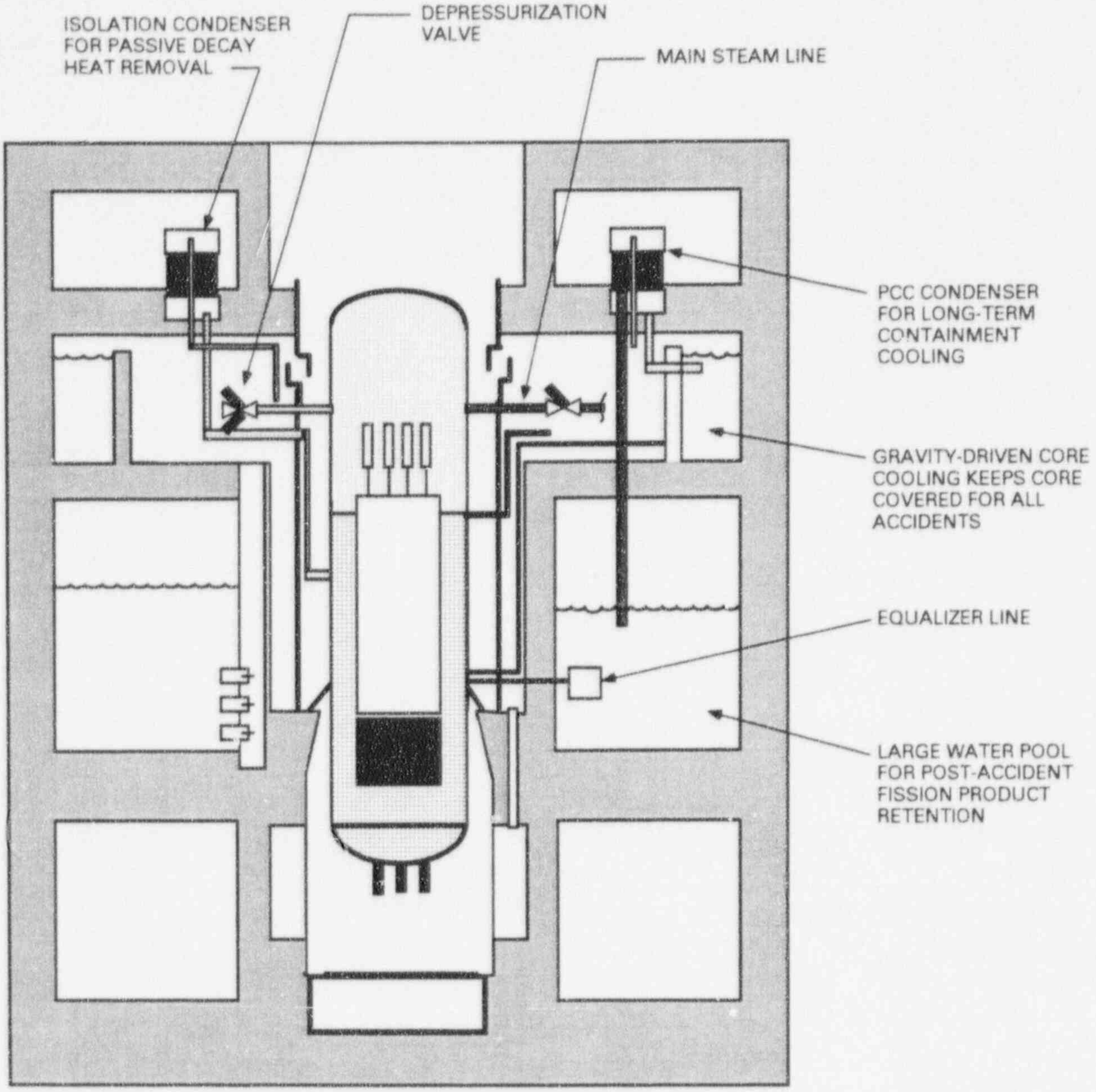


Figure 2.2-1 SBWR Passive Safety Systems

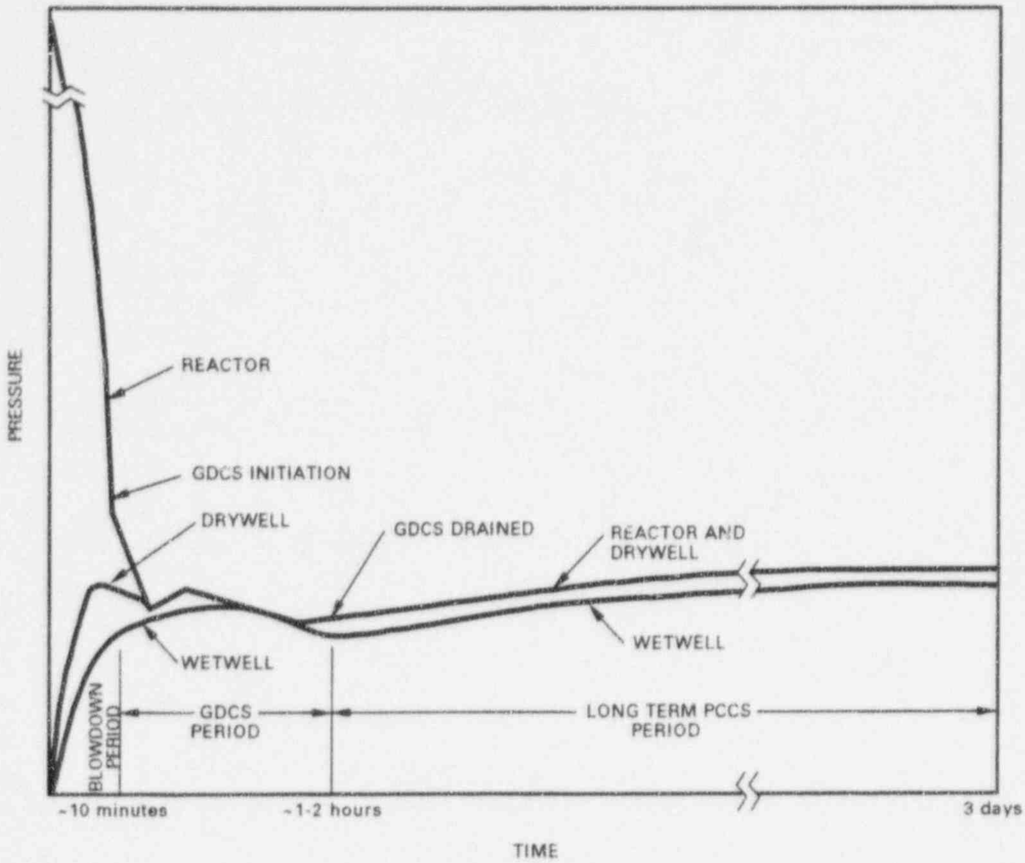


Figure 2.2-2 Phases of the LOCA Transient

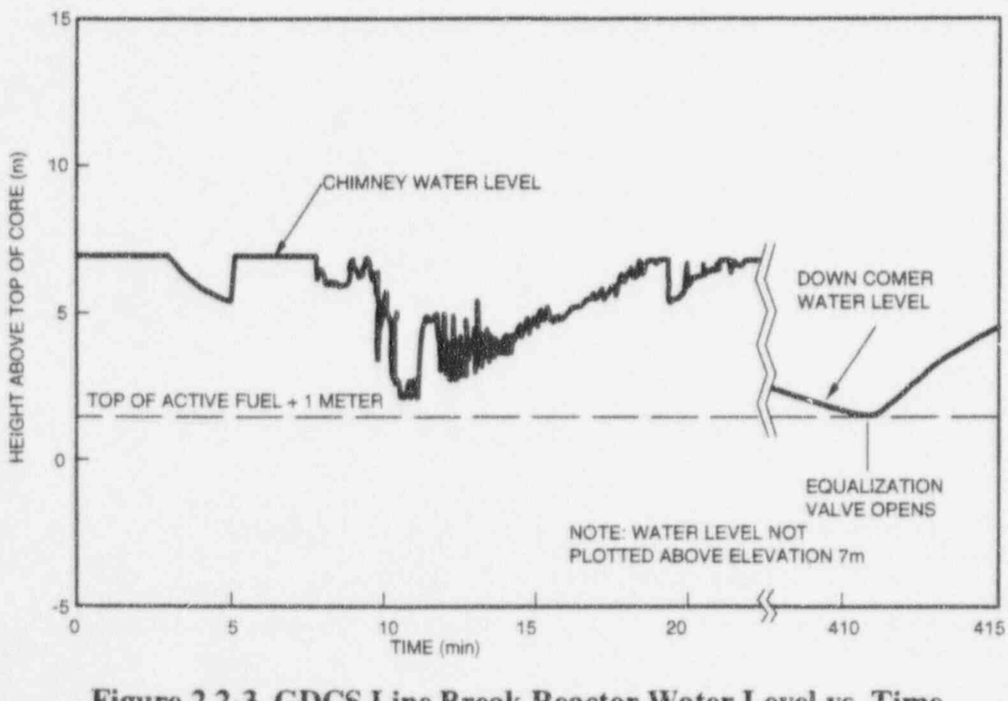


Figure 2.2-3 GDCS Line Break Reactor Water Level vs. Time

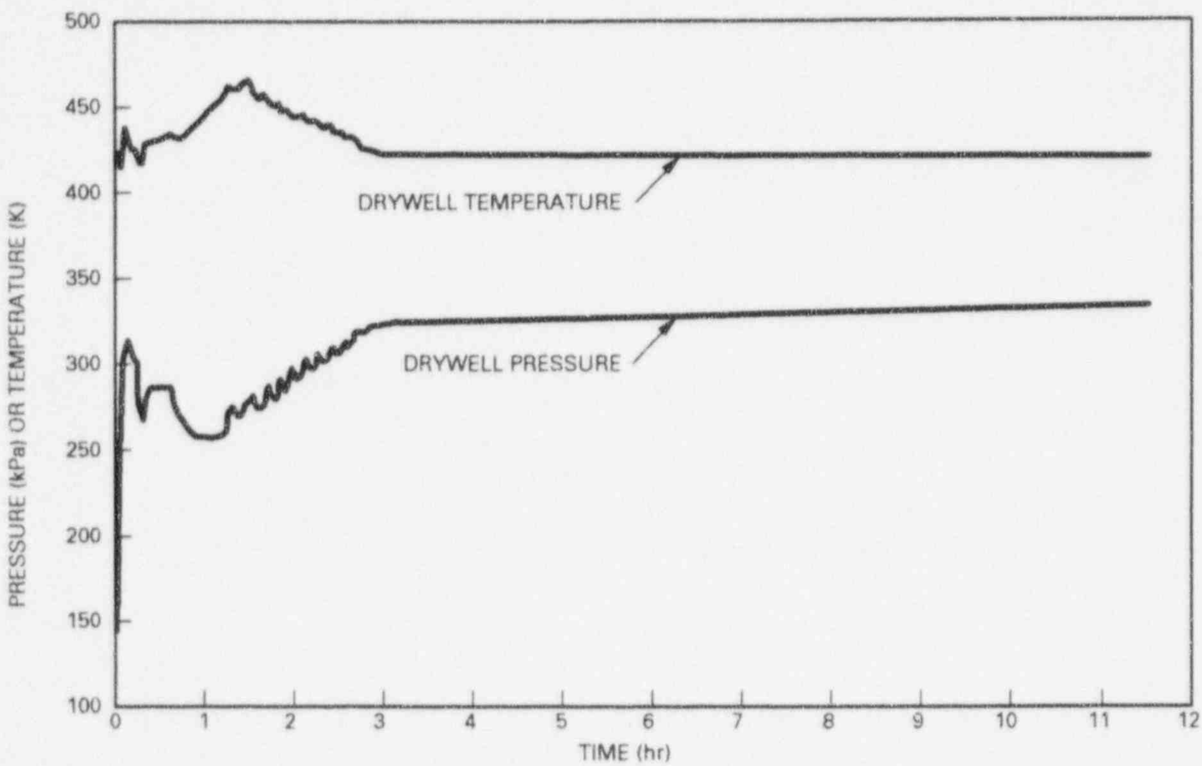


Figure 2.2-4 GDCS Line Break Containment Pressure and Temperature vs. Time

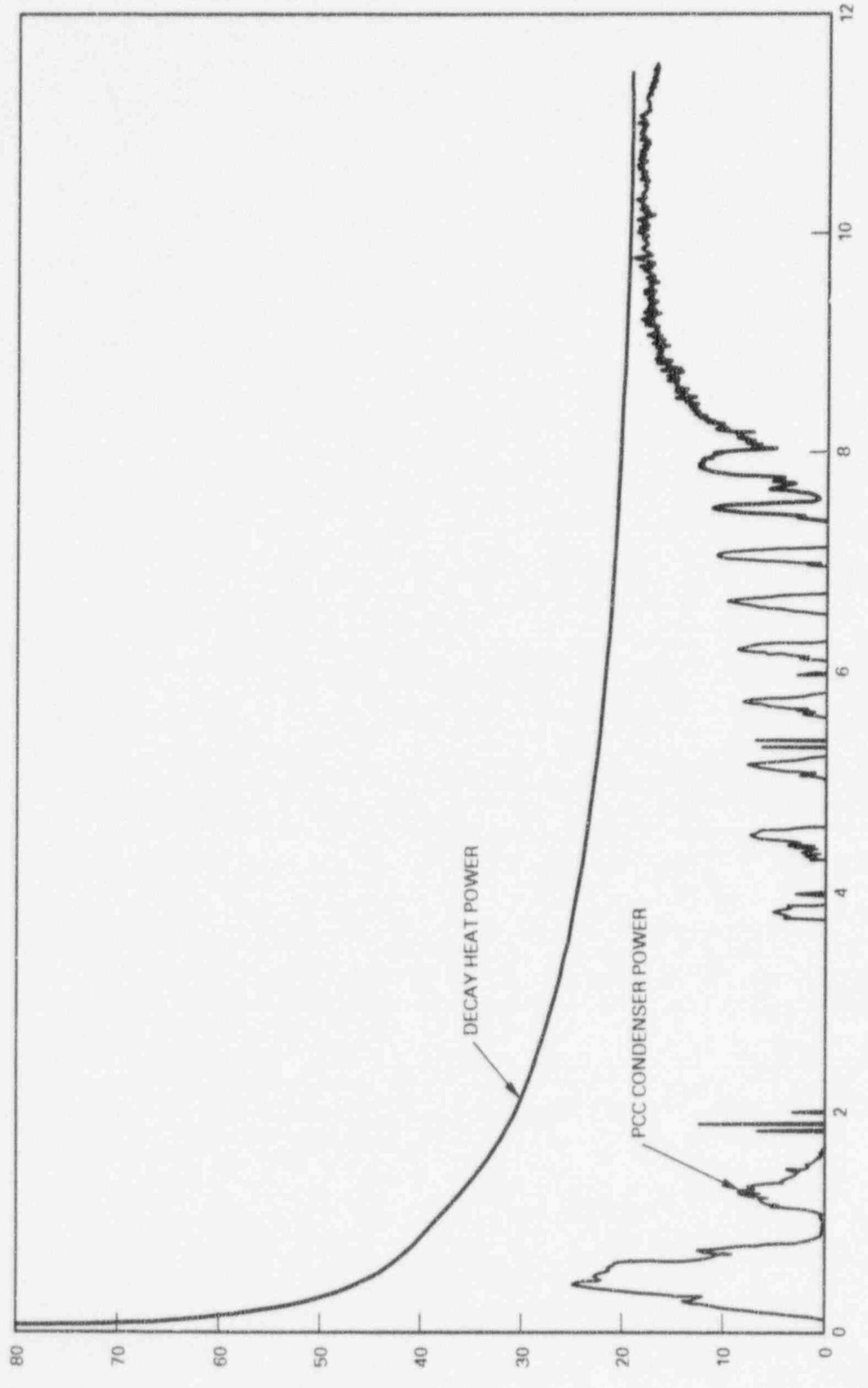


Figure 2.2-5 GDCS Line Break Decay Heat and PCC Power vs. Time

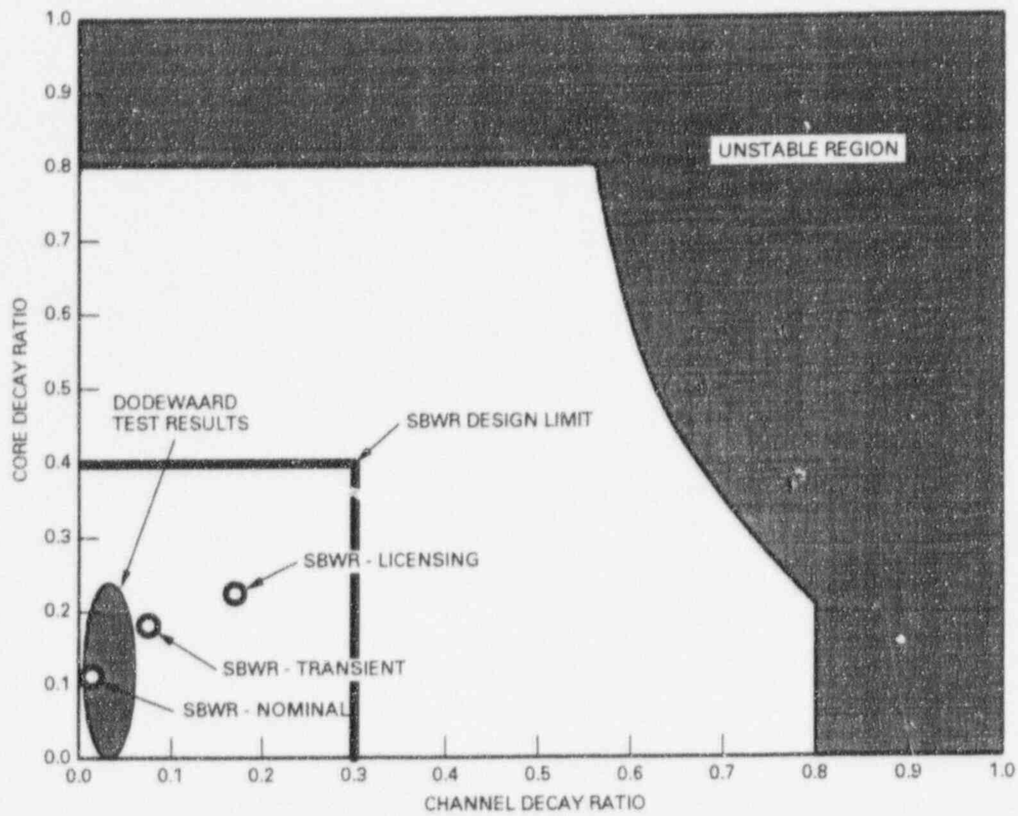


Figure 2.2-6 SBWR Stability Design Criteria and Performance

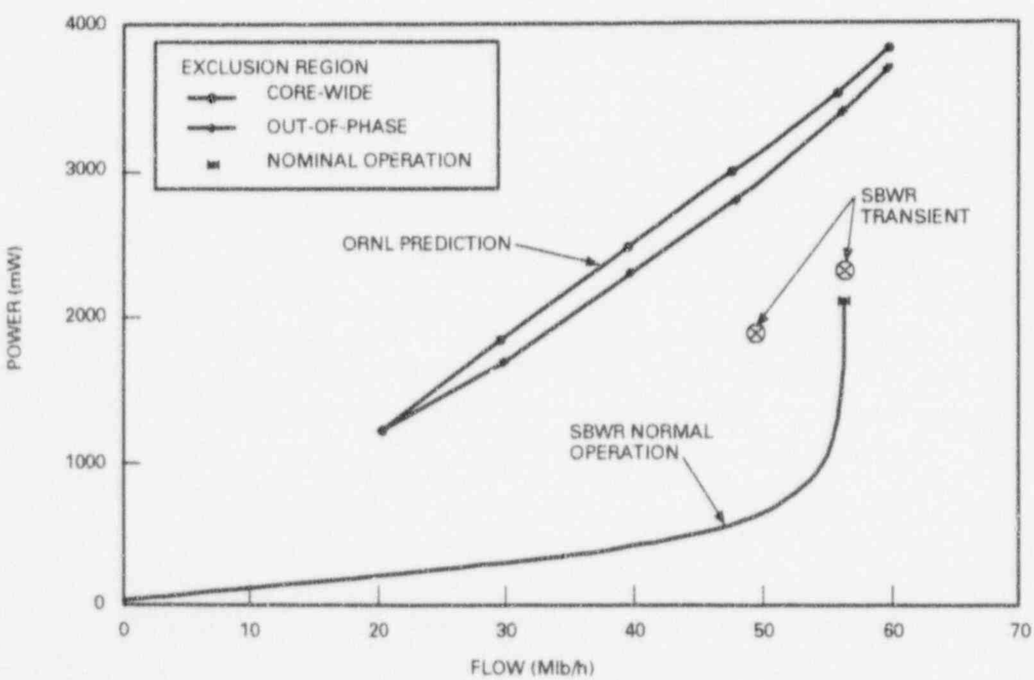


Figure 2.2-7 SBWR Power/Flow Map Comparison With Calculated Stability Limit

2.3 Phenomena Identification and Ranking Tables (PIRT)

The process of Top-Down analysis and qualification of the performance of the SBWR starts with the identification of the important physical phenomena. For this purpose, Phenomena Identification and Ranking Tables (PIRT) [6] were developed. This was done by assembling a team of experts knowledgeable about thermal-hydraulics and transient analysis, and obtaining consensus on the relative importance of various phenomena. Phenomena were given a rank between 0 and 9 based on their "importance" as defined in Section 2.2. The ranking was done on a conservative basis, i.e. generally, phenomena were given a higher rank if there was any uncertainty as to its importance. This resulted in a large number of highly ranked phenomena. It is expected that a much smaller subset will actually prove to be "important" after the tests and sensitivity studies are completed. Tables were developed for small break LOCAs, large break LOCAs, pressurization transients, depressurization transients and reactivity insertion due to cold water injection. Plant startup was also treated as a category of operational transients because of the focus on the potential for geysering. Tables were also developed for ATWS (pressurization events) and for stability during normal operation and transients. In each case, the importance of the phenomena was evaluated for each reactor region: lower plenum, core, upper plenum/chimney, downcomer, etc., as well as for the containment. For the LOCA events, the tables were further subdivided into the blowdown, GDCS and long-term periods of the transients.

It was apparent that for many transients and subregions, the phenomena of importance are the same as for operating BWRs. As an example, for pressurization transients, the most important parameters are the nuclear parameters (void, Doppler and scram reactivity), the interfacial shear (void fraction), subcooled boiling and steam line dynamics. While all these phenomena appear in the PIRT, the phenomena that are unique to SBWR are given primary emphasis in the following sections of this report. These are primarily factors affecting the PCCS performance, GDCS interactions and phenomena associated with natural circulation flow in the core.

The PIRT tables are used for three purposes.

First, the capabilities of the TRACG models are examined to see if all the relevant phenomena can be treated. For this purpose, an evaluation of TRACG models is made with reference to the PIRT parameters, to ensure that all relevant phenomena are modeled. This has been accomplished by verifying that a model with appropriate accuracy exists in TRACG for each phenomenon considered.

Secondly, the qualification database is examined for completeness against the important phenomena. The results of this evaluation are discussed in Chapter 5. Examination of the phenomena ranked "Medium" in importance will be included in Revision C of this document. The medium ranked phenomena will be considered to augment the conservative ranking process adopted by the PIRT team.

Finally, the PIRT is also used in the CSAU process for the determination of model bias and uncertainties. For this purpose, the phenomena ranked "High" in importance will be ranged and sensitivity studies performed to quantify the effect on an appropriate figure of merit. The results from this study will be documented in the Application Methodology Report (NEDE-32178P, Revision 1).

It is recognized that the PIRT is based on engineering judgment. If the planned tests reveal phenomena that were not considered in the development of the PIRT, they will be added to the tables, and their impact on the modeling evaluated.

2.3.1 Loss-Of-Coolant Accident (LOCA)

The overall transient consists of three periods: the blowdown period, the GDCS period and the long-term cooling PCCS period.

For each of these periods, the important thermal-hydraulic phenomena were listed and ranked. This was done by experts familiar with BWR and SBWR characteristics and with transient analysis. The group was interdisciplinary, drawn from several technical areas, such as SBWR design, methods development, and plant transient analysis. The phenomena were classified by reactor and containment region (e.g., lower plenum, core, downcomer, chimney, drywell, wetwell, etc.). Phenomena are ranked separately for small and large breaks. Most of the phenomena and their rankings are similar for small and large breaks. While the front end of the accident progresses more slowly for the small breaks, the rapid depressurization by the Automatic Depressurization System on low water level results in characteristics similar to a large break. The liquid breaks like the GDCS line break and steam line break are not shown separately, but the phenomena important to both have been grouped under "Large Breaks".

2.3.2 Anticipated Transients

Plant startup and three types of operating transients (pressurization, depressurization, and cold water transients) are evaluated. The importance rankings for various phenomena are tabulated by region. "Importance" is ranked by the influence these phenomena have on the Critical Power Ratio (CPR) and maximum pressure reached in the transient. For plant startup, the key criterion is the likelihood of large oscillations in the core flow and power. The nuclear parameters and thermal-hydraulic parameters in the core dominate the pressurization and cold water transients.

2.3.3 Anticipated Transients Without Scram, Stability

These are considered in determining the matrix of tests needed for SBWR performance analysis in Section 5.

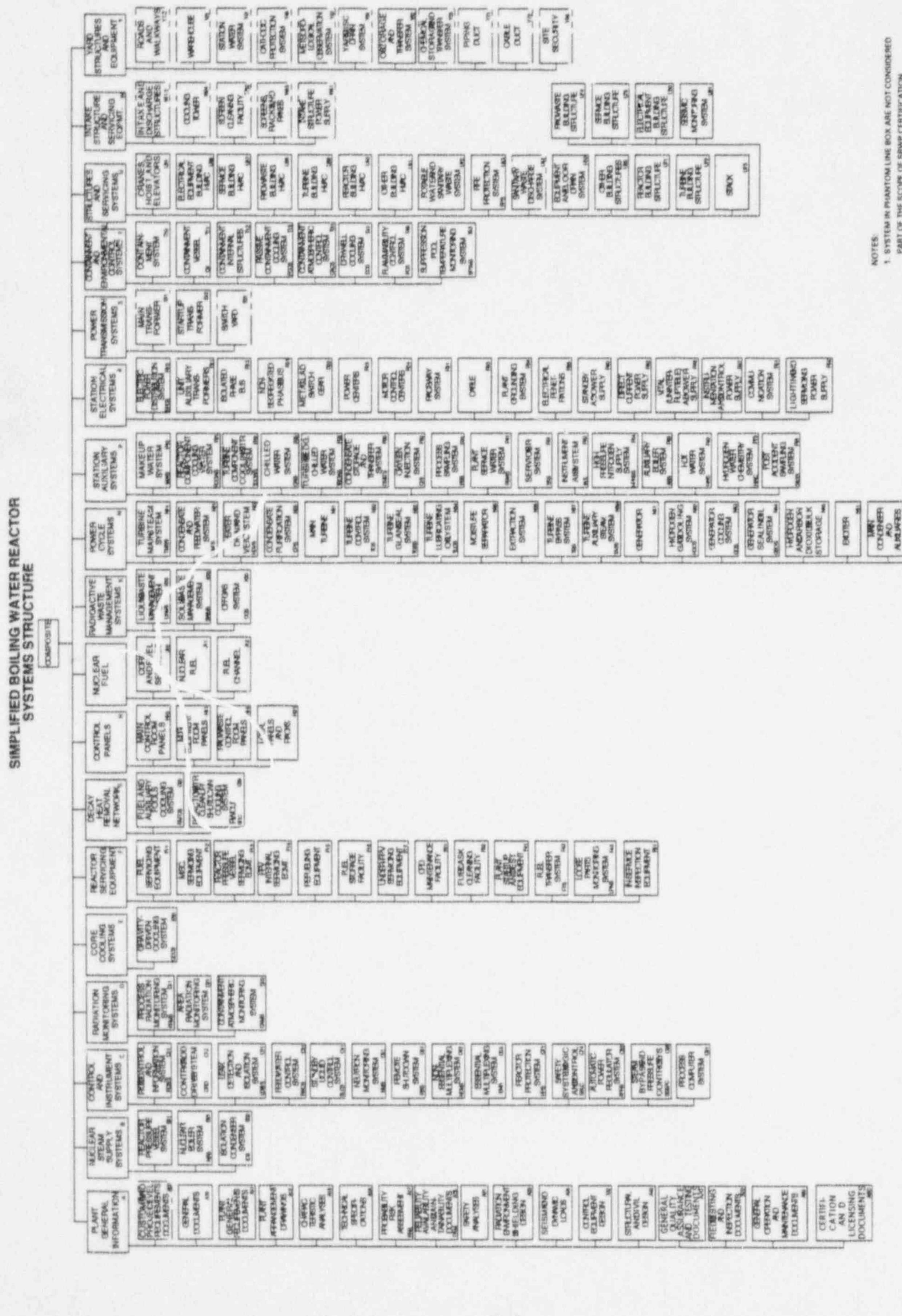
3.0 IDENTIFICATION OF SBWR-UNIQUE FEATURES AND PHENOMENA: BOTTOM-UP PROCESS

3.1 Introduction

This section describes the Bottom-Up process, one of two methods used to develop the test and analysis needs for SBWR. It complements the Top-Down process described in Chapter 2, with which it will be merged in Section 4. This approach compiles a list of SBWR-unique features, associated thermal-hydraulic phenomena and supporting TRACG qualification data. The purpose is to evaluate, from the system and component point of view, the adequacy of the database used to qualify TRACG in the areas important to SBWR thermal-hydraulic response.

3.2 Methodology

Each of the 127 SBWR systems was reviewed to determine if the system was unique or had unique features that do not exist in the BWR operating fleet. Those systems that did not directly affect the thermal-hydraulic response of the SBWR were eliminated. System-unique features, the safety classification of the system, and the MPL number were documented. The principal design engineers were consulted with respect to the current reference system design and unique features, as well as References 3, 31, 32 to determine any new issues associated with that unique feature. For each of the issues, associated important thermal-hydraulic phenomena were identified.



3.2-1 SBWR Product Structure

3.3 Results

A discussion of key results by system is provided in the sections below.

3.3.1 RPV and Internals (B11)

Ten thermal-hydraulic phenomena were evaluated in detail.

3.3.2 Nuclear Boiler System (B21)

Three thermal-hydraulic phenomena were evaluated in detail.

3.3.3 Isolation Condenser System (B32)

Eight thermal-hydraulic phenomena were evaluated in detail.

3.3.4 Standby Liquid Control System (C41)

Five thermal-hydraulic phenomena were evaluated in detail.

3.3.5 Gravity-Driven Cooling System (E50)

Six thermal-hydraulic phenomena were evaluated in detail.

3.3.6 Fuel and Auxiliary Pools Cooling System (G21)

Two thermal-hydraulic phenomena were evaluated in detail.

3.3.7 Core (J-Series)

In the area of the SBWR core, four issues/phenomena were identified as unique to the SBWR.

3.3.8 Containment (T10)

During the review of the SBWR design, 21 unique containment system thermal-hydraulic phenomena were identified.

3.3.9 Passive Containment Cooling System (T15)

The systematic review of the SBWR design identified 13 thermal-hydraulic phenomena related to the design of the

4.0 EVALUATION OF IDENTIFIED PHENOMENA AND INTERACTIONS

The PIRT analysis in Section 2 identified important phenomena for different types of transients and LOCAs. These were grouped by the period of the transient and listed separately for each region of the reactor vessel and containment. In Section 3, a Bottom-Up process was employed to identify SBWR-unique design features and associated phenomena and interactions. These were classified according to the SBWR system (e.g., FAPCS, Nuclear Boiler, etc.) where the particular feature was found. Following the overall strategy described in Section 1.3, the highly ranked phenomena from these lists are now combined in this section to yield a comprehensive, composite list of phenomena that need to be considered. The list is composed of separate tables for phenomena and interactions for each type of transient (LOCA, operational transients, ATWS, etc.). The list of interactions is screened in Section 4.2 and reduced to a final table of phenomena for which data are needed for qualification of TRACG in Section 4.3. In Section 5, these tables will be compared against the Test Plan to confirm that all elements of the tables are covered by tests.

4.1 Composite List of Identified Phenomena and Interactions

These are also picked up by the PIRT. The main additions to the PIRT list came from detailed consideration of the Isolation Condenser units.

4.2 Analytical Evaluation of System Interactions

The purpose of the system interaction study was: (1) to investigate the effects of both active and passive systems which could be available to support Engineered Systems Feature (ESF) systems during a LOCA; and, (2) to determine if interactions between the systems could degrade the performance of the ESF systems from what it would be if they were acting alone. The study extends earlier work presented in Chapter 6 of the SSAR (Reference 3), which evaluated the effect of the break location and of various single failures. A part of this earlier study examined the possible adverse effect of reverse flow through the Isolation Condenser during an inadvertent opening of a DPV. Additional analysis in Chapter 19 of the SSAR (Reference 3) examined use of non-safety grade engineered systems to prevent core damage.

The present study examines both system interactions which could affect the SBWR primary system response, as measured by the fuel temperature and vessel water level, and system interactions which could affect the containment response, as measured by the containment temperature and pressure. The study was performed using the TRACG code with two different input models. System interactions affecting the primary system were studied with the TRACG input model used for LOCA analysis of the SBWR, which provides a detailed representation of the reactor core, vessel internals and associated systems, but a less detailed representation of the containment. For system interactions affecting the containment, the TRACG input model for containment analysis was used. This input model provides a more detailed representation of the containment and its systems but a less detailed reactor pressure vessel model. Both input models have been compared to assure that they predict similar global response behavior of the reactor pressure vessel and containment.

The use of analysis methods is a practical and effective way to evaluate system interactions. The TRACG code and the input models for the primary system and containment which were discussed above include detailed modeling of the important passive and active systems available in the SBWR and can simulate the interactions between these systems during various accident scenarios. This makes it possible to screen a large number of possible system combinations and accident paths to identify those system combinations and accidents most likely to produce adverse interactions. Based on this type of study, final confirmation of interaction effects can then be obtained from integral tests.

4.2.1 Accident Scenario Definition

The systems selected for the study were those that would likely be available during a LOCA and which could produce adverse interactions with the safety grade engineered systems for core and containment cooling.

4.2.2 Results from the Primary Systems Interactions Study

Several different break locations were considered for the primary system interactions study.

4.2.3 Results from the Containment Systems Interactions Study

The containment system interactions study investigated interactions between available safety grade engineered systems as well as interactions of these systems with other systems which could be available for containment cooling without a loss of power.

4.2.4 Summary of System Interaction Studies

The system interactions considered in this study included those considered most likely to occur when some form of external electrical power was available and which were not clearly beneficial to the operation of the safety grade engineered safety systems.

4.3 Summary of Evaluations

This section summarizes the results of screening the phenomena listed in the tables of Section 4.1, primarily in the area of interactions, as a result of the studies of Section 4.2. This constitutes the final step in determining the needs for test data for TRACG qualification. These needs are detailed in Subsections 4.3.1 and 4.3.2 for LOCA and transients, respectively. Subsection 4.3.3 covers ATWS and stability. Section 5 then presents the results of comparing these needs against the test plan.

4.3.1 Transients

All issues but one have been carried forward to Section 5 as needs for TRACG qualification.

4.3.2 ATWS and Stability

For ATWS, the majority of the phenomena are captured either by the Transient PIRT (neutronic and thermal hydraulic issues, Isolation Condenser, etc.) for the reactor parameters or by the Containment PIRT for SRV discharge to the suppression pool (critical flow, pool stratfication and heatup, etc.).

5.0 MATRIX OF TESTS NEEDED FOR SBWR PERFORMANCE ANALYSIS

The tables of important phenomena and interactions from Section 4 were compared with the original Test Plan as it existed when this study began. It was found that most of the identified effects were covered by existing tests which could be used to qualify TRACG. In a few cases, additional testing or qualification was proposed and incorporated into the Test Plan. The resulting matrix of tests needed for TRACG qualification is presented in this section. The tests have been divided into (1) Separate Effects Tests, (2) Component Performance Tests, (3) Integral System Tests, and (4) Operating Plant Data. The first two types of tests are suitable for model development, the latter two for checking the overall performance of the code.

5.1 Separate Effects Tests

The facilities are listed in Appendix A, where the types of tests, test purpose and data available from each are also briefly described.

5.2 Component Performance Tests

A large number of phenomena related to the blowdown and refill processes in the lower plenum, bypass and core are covered by the component tests. Parallel channel effects and separator characteristics are also part of this database.

5.3 Integral System Response Tests

Integral system response tests model overall behavior of a facility subjected to transients simulating specific accidents or transient events. Tests are performed on a scaled simulation of the reactor system.

5.4 Plant Operating Data

The performance of the SBWR is similar to that of other BWRs for operational transients. Plant data are very valuable in validating code performance for complex systems involving an interplay between thermal hydraulics, neutron kinetics and control system response.

5.5 Summary of Test Coverage

The previous sections specified the test facilities and BWR plants from which data have been used (or will be used) for TRACG qualification. This information was tabulated for each of the identified important phenomena, by category of tests (separate effects, component performance, etc.).

6.0 INTEGRATION OF TESTS AND ANALYSIS

This section examines the tasks necessary to complete the qualification of TRACG. Figure 6.0-1 shows the "Road-Map" of how the new and existing test data support SBWR certification.

6.1 TRACG Qualification Plan

Details on the tests and TRACG runs to be performed are identified in Appendix A-Test Plan. The Analysis Plan in Appendix A identifies the specific tests for which blind predictions and post-test analysis will be performed.

6.2 Use of Data for TRACG Model Improvement and Validation

The TRACG computer code is qualified to Level 2 (verified, production) status at GE-NE. Thus, the code configuration is controlled, and the models and the results of validation testing have been reviewed and approved by an independent Design Review Team. In the development process, the separate effects and component data were used for model development and refinement. These data also provided guidelines for the nodalization which was used for all the SBWR calculations. The new data and the results of the post-test analyses will be used in the same way. If changes are necessary to the TRACG models, a new version of the code will be created and brought to a controlled Level 2 status under the GE-NE quality assurance procedures. If changes in the nodalization are indicated, calculations affected by the changes will be redone and reverified.

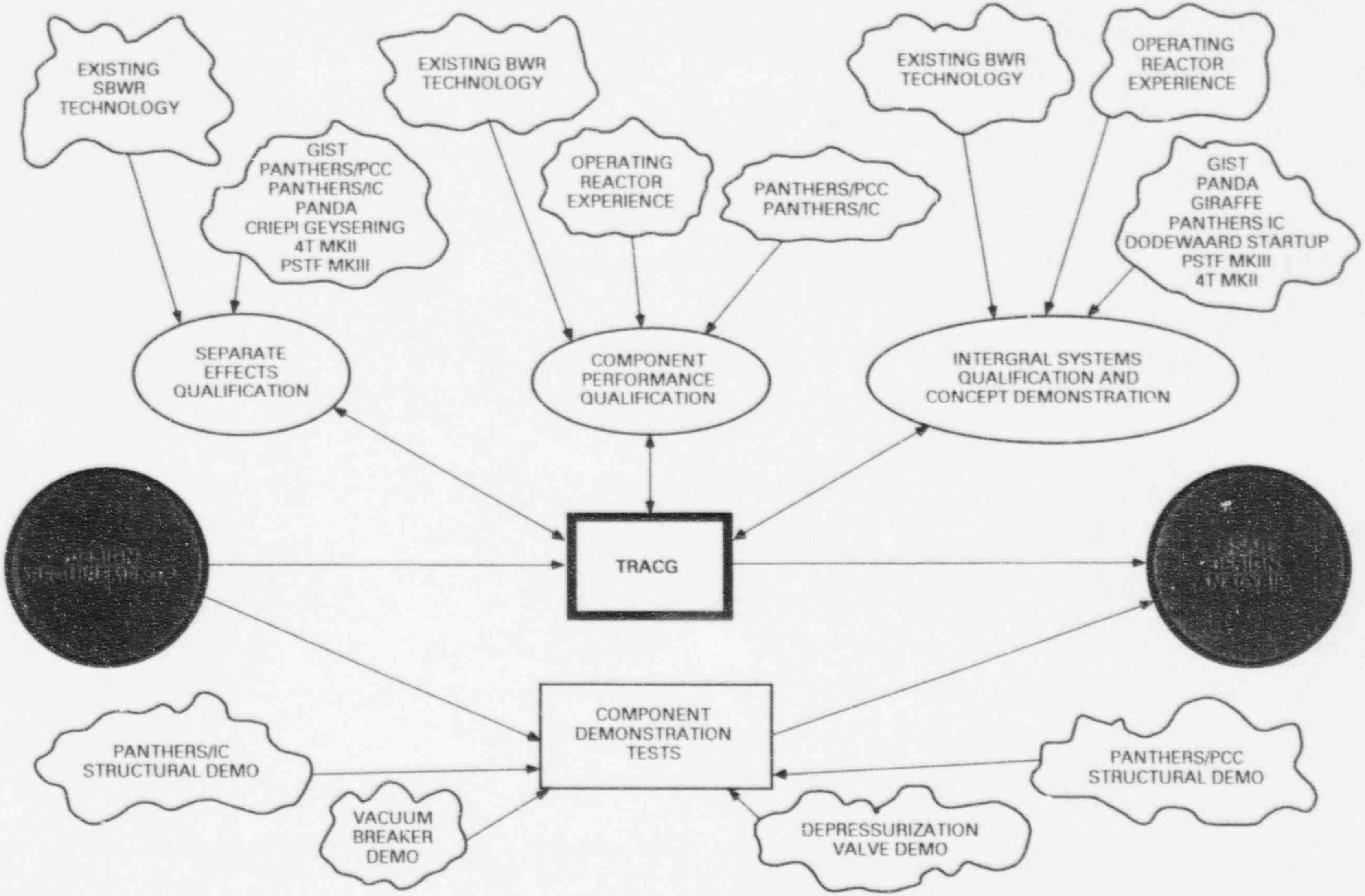


Figure 6.0-1 Technology Basis for SBWR Design

7.0 SUMMARY AND CONCLUSIONS

The Test and Analysis Program Description (TAPD) systematically defined test and analysis needs using Top-Down and Bottom-Up approaches to identify key phenomena, issues and interactions between phenomena and systems (Sections 2, 3, and 4 and Appendix C). These needs were compared to the existing test plan and the existing TRACG qualification plan, and modifications were made where necessary to fill in gaps in the database and the TRACG qualification base (Sections 5 and 6). The Test and Analysis Plan defined the remaining activities for closure (Appendix A). Test facility scaling was addressed quantitatively in Appendix B. This document supersedes previous GE-NE submittals with regard to test objectives, test conditions, data use, and anticipated test analysis.

Several changes in the test and analysis programs resulted from the study documented here. A number of tests were added. In several instances, tasks to be performed have been defined in more detail, and the focus and data usage from some facilities was modified. The following summarizes the key changes:

Test Plan

- GIST: No changes in testing. Data usage focused on GDCS flow and GDCS initiation time.
- GIRAFFE: Phase 1 and Phase 2 data usage changed from primary qualification of TRACG to support use. Helium and systems interaction testing (SIT) added.
- PANTHERS/PCC: No changes in testing or data usage.
- PANTHERS/IC: Test matrix revised to measure performance at lower pressures.
- PANDA: Program added to list of tests required for certification. Test matrix expanded from two to nine transient tests. Program becomes the primary containment and systems interaction data base.

Analysis Plan

- GIST: Analysis Completed.
- GIRAFFE: Helium test and systems interaction test analyses added for TRACG analysis.
- PANTHERS/PCC: Fifteen specific runs identified for TRACG analysis.
- PANTHERS/IC: Six specific runs identified for TRACG analysis.
- PANDA: All six steady-state tests and nine LOCA tests identified for TRACG analysis.
- OTHER TESTS: TRACG analysis of five other tests (1/6 scale Boron mixing, CRIEPI Geysering, PSTF/Mk III, 4T/Mk II, and PSTF Stratification) and one operating plant experience (Dodewaard startup) to address specific identified qualification needs.

The TAPD specifically addresses the requirements of 10CFR52.47 by establishing that a technology basis (a combination of test data, analysis and plant data) exists for the SBWR safety features, for interdependent effects between safety features, and for qualification of the TRACG code used for SBWR safety analysis. Specifically:

- 10CFR52.47 requires that "The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof." The studies summarized in Sections 2, 3 and 4 defined the phenomena important to SBWR safety in two independent ways. These are merged in Section 5 where the testing and experience bases applicable to each are shown. Each important phenomenon is covered by at least one separate effects test, component test, integral systems test, or operating reactor datum.
- 10 CFR52.47 requires that "Interdependent effects among the safety features of the design have been found to be acceptable by analysis, appropriate test programs, experience, or a combination thereof." The studies summarized in Section 4 and Appendix C identified the important interactions. For most of these, analyses or tests already planned suffice to show the effects are negligible or bounded. For a few, additional tests were judged to be necessary. These have been added to the SBWR program.
- 10CFR52.47 requires that "Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analysis over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions." The matrix of tests and operating plant data shown in Section 5 identifies elements which have been used to date (X entries), elements in which existing test data will be used (Q entries), and elements in which forthcoming test data will be used (T,Q entries) to qualify the SBWR analytical model, TRACG. These are collected in Section 6 to show the composite TRACG qualification plan.

GE-NE believes that if the overall TRACG qualification plan described in Section 6, and the SBWR-specific test programs (and associated TRACG analyses) described in Appendix A, are completed with no major surprises, it will be possible to conclude that the provisions of 10CFR52.47(b)(2)(i)(A)(1), (2), and (3) have been satisfied.

8.0 REFERENCES

[1] *TRACG Computer Code Model Description*, J.G.M. Andersen, Md. Alamgir, Y.K. Cheung, L.A. Klebanov, J.C. Shaug. NEDE-32176P, Licensing Topical Report, January 1993.

[2] *TRACG Computer Code Qualification*, J.G.M. Andersen, M.D. Alamgir, J.S. Bowman, Y.K. Cheung, L.A. Klebanov, W. Marquino, M. Robergeau, D.A. Salmon, J.C. Shaug, B.S. Shiralkar, F.D. Shum, K.M. Vierow, NEDE-32177P, Licensing Topical Report, January 1993.

[3] *SBWR Standard Safety Analysis Report*, 25A5113, Rev. A.

[4] *Quantifying Reactor Safety Margins*, NUREG/CR/5249, EGG-2552, Rev. 4.

[5] *Implementation of TRACG for Licensing Analysis*, J.L. Rash to R.C. Jones Jr. (NRC, NRR Chief Reactor Systems Branch), MFN 042-92 and JSC-92-010, March 9, 1992.

[6] *Quantifying Reactor Safety Margins*, B.E. Boyack et. al, Nuclear Engineering and Design (Parts 1-4), 119, Elsevier Science Publishers B.V. (North Holland), 1990.

[7] *Application of TRACG to SBWR Licensing Safety Analysis*, NEDE-32178P, H.T. Kim, February 1993.

[8] *AEOD Concerns Regarding The Power Oscillation Event at LaSalle 2 (BWR-5)*, USNRC, AEOD Special Report S803, 1988.

[9] *Leibstadt Stability Tests During Startup Testing* (contained in Reference 5).

[10] *Stability Investigations of Forsmark-1 BWR*, B. Anderson, et. al., International Workshop on BWR Stability, October 1990.

[11] *Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report*, NEDE-25445, August 1982.

[12] *ODYSY01/ODYSY02 Qualification Report*, NEDE-30227, (includes Peach Bottom-2 Stability Test Data), July 1983.

[13] *The Startup of the Dodewaard Natural Circulation BWR – Experiences*, W.H.M. Nissen et. al., N.V. GKN, Netherlands, Waalbandijk 112a, 6669 MG Dodewaard (ANP '92), Tokyo, Japan (Detail measurements are contained in GKN-report 92-017/FY/R), October 25-29, 1992.

[14] *Stability Monitoring of a Natural-Circulation-Cooled Boiling Water Reactor*, T.H.J.J. van der Hagen, Thesis, Delft Univ. of Technology, The Netherlands, 1989.

[15] *Measurements at Various Pressures at the Dodewaard Natural Circulation Boiling Water Reactor in Cycle 23*, T.H.J.J. van der Hagen, GKN-Report 93-023/FY/R, 1993.

[16] *Hatch Unit 2 Two-pump Trip Test*, November 18, 1978 (data contained in Reference 5).

[17] *Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36 Rod BWR Fuel Element with Non-Uniform Axial and Radial Heat Flux Distribution*, O. Nylund, et. al., FRIGG-4, ASEA-ATOM, December 1970.

[18] *ABWR Horizontal Vent Containment Tests*, NEDC-31393 - SS Test Series.

[19] *MIT and UCB Separate Effects Tests for PCCS Tube Geometry, "Single Tube Condensation Test Program"*, NEDC-32301.

[20] *Pressure Suppression Test Facility (PSTF)*, NEDE-13377.

[21] *MSC/NASTRAN Manual Version 67* Macneal-Schwendler Corporation, Los Angeles, CA.

[22] *ABWR Horizontal Vent Containment Tests*, NEDC-31027 – FS series.

[23] *Steam Purge Clearing Tests*, NEDE-28853.

[24] *1/3 Area Scaled – Single Vent Row Tests*, NEDE-21596P.

[25] *1/9 Area Scaled – 3 Vent Row Tests*, NEDE-24720P.

[26] JSBWR Phase 2 Report, EBWR and VK50 Tests

[27] CRIEPI Tests, Thermo-Hydraulic instability of natural circulation BWRs (Explanation on instability mechanisms at startup by homogeneous and thermodynamic equilibrium model considering flashing effect.) by Fumio Inada and Tomio Ohkawa, Komae Research Laboratory, Central Research Institute of Electric Power Industry (CRIEPI). Paper presented at 1994 Conference in Italy.

[28] *BWR 5 - 1/6 scale Boron Mixing Tests*, NEDE-22267.

[29] *ABWR - 1/6 scale Boron Mixing Tests*, NEDC-30326.

[30] Document No. BN-TOP-3, Bechtel Power Corp, San Francisco, CA, August 1975.

[31] *Design and Analysis Similarities Between the ABWR and SBWR*, NEDC-32231, December 1993.

[32] *Scaling of the SBWR Related Tests*, NEDC-32288, G. Yadigaroglu, July 1994.

[33] *Sixteen-Rod Heat Flux Investigation, Steam-water at 600 to 1250 psia*, E. Janssen, GE.

[34] *The Effects of Lengths and Pressure on the Critical Heat Flux for a Closely Spaced 19-Rod Bundle in Forced Convective Boiling*, Dept. of Chem Eng., Engineering Research Lab Heat Transfer Research Facility, Columbia University, NY.

[35] *NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design*, Letter MFN No. 078-94 from P.W. Marriott (GE) to Richard W. Borchardt (USNRC), March 31, 1994.

[36] *Quantifying Reactor Safety Margins — Application of Code Scaling Applicability and Uncertainty Evaluation Methodology for a Large-Break Loss-of-Coolant Accident*, B. Boyack et al., NUREG/CR-5249, 1989.

[37] (See Reference 7)

[38] *Mark II Pressure Suppression Test Program Phase II and III Tests*, NEDE-13442P-01, May 1976. *Mark II Pressure Suppression Test Program Phase II and III Tests*, NEDE-13468, October 1976.

[39] *Hierarchical, Two-Tiered Scaling Analysis, Appendix D to An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution*, Nuclear Regulatory Commission Report, NUREG/CR-5809, EGG-2659, November 1991.

[40] GE COMPASS database, run dated July 1994.

[41] GE Master File Number MFN 042-92, Letter J.L. Rash (GE) to R.C. Jones (NRC), *Implementation of TRACG for Licensing Analysis*, March 9, 1992.

[42] GIST Final Test Report, *SBWR Program Gravity-Driven Cooling System Integrated Systems Test*, GEFR-00850, October 1989.

[43] *GIRAFFE Passive Heat Removal Testing Program*, by K.M. Vierow, NEDC-32215P, June 1993.

[44] *Test Report for SBWR Depressurization Valve Operational and Flow Rate Test*, WYLE Report #41152-0, September 1990.

[45] *Startup of the Dodewaard Natural Circulation Boiling Water Reactor*, Hagen, T.H.J.J. van der, Karuza, J., Nissen, W.H.M., Stekelenburg, A.J.C., Wouters, J.A.A., GKN Report 92-017/FY/R, 1992.

[46] *Full-Scale Mark III Tests, Test Series 5707*, NEDE-21853-P, August 1978.

[47] *Mark III Confirmatory Test Program, 1/3 Scale Condensation and Stratification Phenomena, Test Series 5807*, NEDE-21596P, March 1977.

[48] *A Comparison of the RELAP Simulation of Mark III Suppression Pool Thermal Stratification with Data from the Pressure Suppression Test Facility*, NEDE-21957P, September 1978.

[49] *Transient Behavior of Natural Circulation for Boiling Two-Phase Flow (2nd Report: Mechanism of Geysering)*, M. Aritomi, et al., Transactions of First JSME/ASME Joint International Conference on Nuclear Engineering (ICONE-1), Tokyo, Japan, pp. 87-94, November 4-7, 1991.

[50] *PANTHERS Test Plan & Procedure*, SIET document No. 0098 PP 91, Revision 1.

[51] *Technical Specification For IC & PCC Instruments Installation*, SIET document No. 00157 ST 92, Revision 1, January 1, 1994.

[52] *PANTHERS-PCC Test Facility Instrumentation, Data Acquisition, & Processing Specification*, SIET document No. 00095 RS 91, Revision 1, June 8, 1994.

[53] *Isolation Condenser and Passive Containment Condenser Test Requirements*, GE Nuclear Energy, 22A6999, Revision 3.

[54] *PANDA Test Specification*, GE Nuclear Energy, Revision 1.

[55] *PANDA Steady State PCC Performance Tests, Test Plan and Procedures*, Paul Scherrer Institut, document No. ALPHA-410, February 1995.

[56] *PANDA Pre-Test Analysis*, NUCON Report 40315-NUC-94-7034, GE Nuclear Energy Letter MFN 119-94.

[57] *GIRAFFE Test Specification*, GE Nuclear Energy, 25A5677, Revision 0.

[58] *Guide for Quality Assurance of Nuclear Power Plants*, Electrotechnical Standard Survey Committee, Japan Electric Association, JEAG 4101-1990.

[59] *PANTHERS Pre-Test Calculation* (contained in Reference 35).

[60] *Thermally Induced Flow Instabilities in Two-Phase Mixtures*, Ishii, M., and Zuber, N., 4th International Heat Transfer Conference, Paris, Paper No. B5.11, 1970.

[61] *Drift Flux Model for Large Diameter Pipe and New Correlation for Pool Void Fraction, Kataoka*, I. and Ishii, M., Intl. J. Heat Mass Transfer, vol. 30, 1927-1939, 1987.

[62] Scaling and Analysis of Mixing in Large Stratified Volumes, Peterson, P.F., Intl. J. Heat Mass Transfer, vol. 37, 97-106, 1994.

[63] *The Thermal Hydraulics of a Boiling Water Reactor*, Lehey, Jr.,R.T., Moody, F.J., American Nuclear Society, 1977.

[64] *Thermo-hydraulic Instability of Natural Circulation BWRs at Low Pressure Start-up: Experimental Estimation of Instability Region with Test Facility Considering Scaling Law*, Fumio INADA, Masahiro FURUYA and Akira YASUO, Central Research Institute of Electric Power Industry, CRIEPI, Hiroaki TABATA and Yuzuru YOSHIOKA, Japan Atomic Power Company, JAPC, H. T. Kim, GE Nuclear Energy, April 1995.

[65] *TRACG Analyses of Flashing Instability During Start-up*, Andersen, J.G.M. and Klebanov, L.A., GE Nuclear Energy, April 1995.

[66] Thermal-Hydraulic Oscillations in a Low Pressure Two-Phase Natural Circulation Loop at Low Powers and High Inlet Subcooling, S.B. Wang, J.Y. Wu, Chin Pan and W.K. Lin, National Tsing Hua University, 4th International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety, April 6-8, 1994.

APPENDIX A — TEST AND ANALYSIS PLAN (TAP)

A.1 Introduction

This appendix identifies the specific tests and analyses that will be performed to meet the identified supplemental needs.

The goal of the SBWR Test Program is to provide a sufficient data base to support certification of the SBWR as a standard design. Consequently, the scope of the test program goes beyond establishment of the TRACG qualification database, in that demonstration testing of concepts unique to the SBWR, or equipment having design requirements not previously analyzed or tested, is also included. This testing is also described in this appendix. In many cases, the same test data are used for both applications.

Section A.2 provides an overview of the philosophy used in determination of specific tests and analyses, definition of test types, and an overview of the test effort. Section A.3 presents the Test and Analysis Plan. The following information is provided for each identified test:

Test Plan

- A test description including overviews of test facilities, instrumentation, and procedures.
- Objectives for each test program (and specific tests, as applicable).
- Test matrices, cross referenced to the test objectives, and descriptions of how the data will be used to meet the test objectives.
- Justification of the test conditions.

Analysis Plan

- Test runs identified for TRACG analysis
- Description of how the identified comparisons between test and analysis meet the qualification needs

This document supersedes previous submittals with regard to test objectives, test conditions, data use, and anticipated test analysis.

A.2 Test and Analysis Philosophy

A.2.1 Test Types

The overall goals of the SBWR Test and Analysis Program are to be met by several types of testing, in several different facilities, world wide. Testing is divided into:

- ***Thermal-Hydraulic Testing*** — provides data necessary for qualification of TRACG and for demonstration of the concepts of passive safety systems design. Thermal-hydraulic testing is further subdivided into (1) steady-state and separate effects tests, (2) component performance tests, (3) integral systems tests, and (4) concept demonstration tests.
- ***Component Demonstration Testing*** — provides data on the capability of specific equipment to meet its design objectives.

A.2.2 Test Overview

SBWR thermal-hydraulic testing is summarized in Table A.2-1. The test program consists of 124 steady-state test conditions, 15 transient performance demonstrations, and 45 integral systems tests. Subsection A.3.1 describes each of the four facilities (PANTHERS, PANDA, GIST, and GIRAFFE) in which these tests will be or have been performed, and includes specific test objectives, test matrices and descriptions of how each of the test groups addresses the test objectives.

Subsection A.3.1.8 also gives an overview of other data that will be used for TRACG qualification beyond the qualification described in Reference 2.

SBWR component performance tests are described in Subsection A.3.2, including testing of the PCC and IC heat exchanger components, depressurization valves (DPVs), and vacuum breaker valves (VB).

A.2.3 Test Approach

The philosophy of testing is to focus on those features and components that are SBWR-unique or performance-critical, and to test over a range that spans and bounds the SBWR parameters of importance. In general, TRACG is used to predict the SBWR parameter range for the spectrum of accidents and transients, and then that range is bounded in the test matrix. Some SBWR tests are performed in a scaled configuration. For these tests, the values of the important parameters are scaled to be consistent with this philosophy. This approach is discussed in Reference 32 and Appendix B.

Additionally, it is the program philosophy to test in multiple scales wherever possible. In these cases, initial conditions for the various tests have been made as similar as possible. Multiple scale testing is useful, since it validates the scaling approach and allows a better understanding of the thermal-hydraulic phenomena involved.

A.2.4 Analytical Approach

The analytical approach to be used is consistent with that previously documented in the TRACG Qualification Licensing Topical Report, (Reference 2). Briefly, the approach is to choose a representative sampling of test data which comprise separate effects, component performance, and integral systems effects, and to perform either pre-test or post-test analysis using TRACG. Tests are chosen for analytical prediction based on their adequacy to demonstrate model prediction capability over the range of predicted SBWR conditions. Sufficient tests are chosen from certification data to establish model adequacy. Additional tests have been chosen from supporting data to confirm the certification predictions, over a wider range of test conditions, or at intermediate points.

It is planned to produce a number of "double blind" pre-test analyses for those certification data experiments not yet performed. Double blind indicates that the analyst has no information on either the results or the exact initial conditions of the experiments. These predictions are based on the as-designed facility configurations, and will be verified.

Following completion of individual tests, additional test runs will be analyzed with TRACG and compared with the test results. These post-test analyses will be performed with the analyst having knowledge of the test results, but will utilize the same nodalization and modeling as the "double blind" predictions, corrected, if necessary, to reflect facility as-built geometry and the actual initial conditions. The objective is to establish the adequacy of the TRACG model in this application. All input decks will be verified.

TRACG modeling or nodalization changes are not expected, but will be made if deemed necessary following an assessment of TRACG predictive capability.

A.2.5 Documentation of Tests and Analysis

A.2.5.1 Test Documentation

Testing is documented by submittal of a series of reports and other documentation that define the configuration of each SBWR test facility, the evaluations performed in conducting the tests, and the results of the testing. Table A.2-2 provides a listing of these submittals. In those cases where the documentation has already been submitted, Table A.2-2 also includes the submittal date, and a reference document identification.

Tables-of-Contents for those report types that apply to all four major test programs [Apparent Test Results (ATRs), Data Transmittal Reports (DTRs) and Data Analysis Reports (DARs)] are included as Attachment A1. In addition, an SBWR Test Program Licensing Topical Report will be submitted summarizing the results from all testing, and integrating the findings from all of the test programs.

A.2.5.2 Analysis Documentation

The results of the TRACG analysis will be documented in the form of pre-test predictions for selected tests, preliminary validation results for each set of tests, and a final TRACG Qualification Licensing Topical Report. The Licensing Topical Reports are listed in Section 1.4.2. The other analysis reports that are planned for submittal to the NRC are listed in Table A.2-3.

Each of the preliminary validation reports will include the objective of the qualification task, the use of the data, a description of the TRACG model and a discussion of the results. The proposed Table of Contents for the Preliminary Validation Results documents is shown in Attachment A1. The results of these post-test calculations as well as other supporting qualification studies will be integrated into the final Licensing Topical Report (LTR), entitled "TRACG Qualification for SBWR". This LTR will supplement the previous LTR on TRACG Qualification, and will include comparisons with data from all the SBWR-specific facilities. This LTR will discuss the overall strategy, nodalization of the reactor vessel as well as the containment, and the evaluation of model uncertainties and bias for SBWR application. The detailed Table of Contents is provided in Attachment A1.

A.3 Test and Analysis Plan

A.3.1 Thermal-Hydraulic Tests

A.3.1.1 PANTHERS/PCC

A.3.1.1.1 Test Description

Overview

PANTHERS/PCC (Passive Containment Condenser) testing is performed as a joint effort by GE, Ansaldo, ENEA, and ENEL at Societa Informazioni Esperienze Termoidrauliche (SIET) in Piacenza, Italy. The test facility consists of a prototype PCC unit, steam supply, air supply, and vent and condensate volumes sufficient to establish PCC thermal-hydraulic performance. Both thermal-hydraulic and component structural demonstration tests are performed in this facility. This section covers the thermal-hydraulic portion of the testing; component structural performance tests are covered in Subsection A.3.2.1.

The PCC condenser is a full-scale, two-module vertical tube heat exchanger designed and built by Ansaldo. Figure A.3-1 is an outline drawing of the heat exchanger assembly. It should be noted that the heat exchanger is a prototype unit, built to prototype procedures and using prototype materials. Three heat exchanger units (6 modules) would be found in an SBWR. The PCC is installed in a water pool having the appropriate volume for one SBWR PCC assembly.

Instrumentation

Figure A.3-2 is a schematic of the PANTHERS/PCC facility. The primary instrumentation specified is sufficient to ascertain heat exchanger thermal-hydraulic performance by performing mass and energy balances on the facility. Additionally, four heat exchanger tubes are instrumented in such a way that local heat flux information may be obtained.

All test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. Table A.3-1 defines the thermal-hydraulic measurements taken during the PCC tests. Additional information may be found in the PANTHERS/PCC Test Plan and Procedure (Reference 50), the Technical Specification of IC and PCC Instrument Installation (Reference 51), the PANTHERS PCC Test Facility Instrumentation, Data Acquisitions, and Processing Specification (Reference 52), and the Isolation Condenser and Passive Containment Condenser Test Requirements (Reference 53).

Test Method

The majority of the PANTHERS/PCC testing is steady-state performance testing. For these tests, the facility is placed in a condition where steam or air/steam mixtures are supplied to the PCC, and the condensed vapor and vented gases are collected. All inlet and outlet flows are measured. The condensate is returned to the steam supply, and the vented gas is released to the atmosphere. Once steady-state conditions are established, data are collected for a period of approximately 15 minutes. The time-averaged data are reported and analyzed.

Steady-state tests using a steam/air mixture are performed as follows. The test loop and PCC condenser are first purged with steam to remove any residual air from the system and to heat the PCC pool to saturation. When the pool is boiling, the required steam flow rate is established, followed by establishment of the required air flow rate to the PCC. The desired PCC inlet pressure is then established by adjusting the position of the vent tank flow control valve. When steady conditions have been established, data is taken for a period of approximately 15 minutes.

A slightly different procedure is used for the steam-only tests. In this case, the vent tank is isolated by installation of a blind flange on the vent line. Following purging of the system, the desired steam flow rate is established. The inlet pressure is not controlled, but allowed to stabilize while maintaining full condensation at the desired steam flow rate. Again, data is then acquired for a period of approximately 15 minutes.

PANTHERS/PCC transient condenser performance tests are used to establish noncondensible buildup effects and PCC pool water level effects. They are not intended to be integral systems tests.

The noncondensible build-up tests are performed as follows. The test conditions are initialized, using the steam-only procedure described in the steady-state test section. When steady-state conditions are established, the data acquisition system is started, and air, helium, or an air/helium mixture is injected at the rate specified. The inlet pressure is allowed to increase as the noncondensibles collect in the vent tank, and the condensation process is degraded by the presence of noncondensibles in the PCC heat exchanger. The test is terminated when the PCC heat exchanger reaches its design pressure.

For the water level tests, the procedure is to establish the initial conditions as described in the steady-state air/steam mixture tests, then to initiate data acquisition. With the position of the vent flow control valve fixed, the PCC pool water level is allowed to decrease by either boil-off, draining, or a combination of the two. Inlet pressure to the PCC is allowed to rise, consistent with the condensation process. The test is concluded when the desired water level range has been investigated.

A.3.1.1.2 Test Objectives

The test objectives of the PANTHERS/PCC Test Program are:

1. Demonstrate that the prototype PCC heat exchanger is capable of meeting its design requirements for heat rejection. *(Component Performance)*
2. Provide a sufficient database to confirm the adequacy of TRACG to predict the quasi-steady-heat rejection performance of a prototype PCC heat exchanger, over a range of air flow rates, steam flow rates, operating pressures, and superheat conditions, that span and bound the SBWR range. *(Steady-State Separate Effects)*
3. Determine and quantify any differences in the effects of noncondensible buildup in the PCC heat exchanger tubes between lighter-than-steam and heavier-than-steam gases. *(Concept Demonstration)*

A.3.1.1.3 Test Matrix and Data Analysis

Steady-State Performance Tests

Table A.3-2a shows the PANTHERS/PCC Steady-State Performance Matrix for Steam-Only Tests. Thirteen test conditions are included.

- Test Conditions 37 through 43 (Test Group P1) are used to determine the baseline heat exchanger performance over a range of saturated steam flow rates without the presence of noncondensible gases. Test Group P1 data are compared with design requirements to meet Test Objective 1. Test Conditions 44 through 49 (Test Group P2) address the effect of superheat conditions in the inlet steam. Test Conditions 38, 44, 45, and 46 may be used to establish the effects of superheat at a relatively low steam flow condition, while Test Conditions 41, 47, 48, and 49 will give the same information at a steam flow rate near rated conditions.

Table A.3-2b shows the PANTHERS/PCC Steady-State Performance Matrix for Air/Steam Mixture Tests. As noted previously, the independent variables are steam mass flow rate, air mass flow rate, steam superheat conditions, and absolute operating pressure. Figure A.3-4 shows the relationship between the steam and air flow rates specified for PANTHERS/PCC testing and the SBWR expected range.

- Test Conditions 9, 15, 18, and 23 (Test Group P3) will be used to compare heat rejection rates over a range of air flow rates to the saturated, steam-only condition determined from Test Condition 41 in the pure steam series. Holding steam flow constant at near rated conditions, these tests yield the effect of air on the condensation process.
- Test Conditions 2, 13, 16, 17, 19, 22, and 25 (Test Group P4) supplement Test Group P3, in that they define condensation performance at the extremes of the SBWR air/steam mixture ranges, and at several intermediate points. These tests will be used to quantify noncondensible effects at off-rated conditions. They will be compared to the appropriate Test Conditions in the P1 group.
- Test Conditions 35 and 36 (Test Group P5) further supplement Test Group P4 by extending the effect of noncondensible gases over the superheated steam range. These tests can be compared to Test Conditions 48 and 49 to establish the effect of air content at the same superheat condition, and to Test Condition 23 at the same air flow, but with saturated steam.
- Test Conditions 1, 3, 4, 5, 6, 7, 8, 10, 11, 12, 14, 20, 21, and 24. (Test Group P6) are lower priority tests. They are run at only a single inlet pressure to supplement the previously identified tests by increasing the data density within the already established air/steam flow map.

Transient Test Conditions

Table A.3-2c shows the PANTHERS/PCC Noncondensible Buildup Test Matrix. Eight test conditions are specified as Test Group P7. In these tests, steam is supplied at a constant rate, and steady-state conditions are established in a manner similar to that of the steady-state performance tests. Air, helium, or air/helium mixtures are then injected into the steam supply, with the vent line closed. The transient degradation in heat transfer performance will be measured, as a function of the total noncondensible mass injected.

- Tests Conditions 50 and 51 provide a baseline condition with air as the only noncondensible. Air is similar to nitrogen in molecular weight, and is heavier than steam. Test Conditions 52 and 53 are similar to Test Conditions 50 and 51, but with the steam supply superheated. Test Conditions 75 and 76 repeat Test Conditions 50 and 51, but utilize helium as the noncondensible gas instead of air. Helium is lighter than steam, and will mix in a manner similar to hydrogen. The results of Tests 50 through 53 plus 75 and 76 can be compared to establish performance differences between lighter-than-steam and heavier-than-steam gases as they build up in the heat exchanger tubes. Test Conditions 77 and 78 can be used to evaluate the effect of an air and helium mixture concurrently flowing into the heat exchanger.

- Test Group P7 data will be evaluated to meet the requirements of Test Objective 3.

Table A.3-2d shows the PANTHERS/PCC Pool Water Level Effect Test Matrix. Three test conditions are specified as Test Group P8. In these tests, steam and air/steam mixtures are supplied to the PCC heat exchanger, and steady-state conditions established in a manner similar to the steady-state performance tests. In these tests, however, the water level in the PCC pool is allowed to drop and the PCC tubes to uncover. Both the PCC pool level and the PCC heat rejection rate are monitored as a function of time.

- Test Conditions 54, 55, and 56 establish the effect of water level in the PCC pool for a range of steam and air/steam supply rates to the PCC. Data from Test Conditions 54, 55, and 56 can be compared to Test Conditions 41, 15, and 25, respectively, to obtain the effect of lowered water level on condensation performance. Test Conditions 54 and 55 can be compared to establish the effect of air content on the rate of pool boiloff.

- Test Groups P1 through P5, P7 and P8 provide a database for TRACG qualification and meet test objective 2.

A.3.1.1.4 Justification of Test Conditions

PCC Operational Modes

In the SBWR, the post-LOCA function of the PCC heat exchanger is to remove decay heat from the drywell and reject this energy to the atmosphere. This is the major difference between the SBWR and earlier pressure suppression containment designs. In earlier designs, the decay heat is transferred from the drywell to the wetwell via the main vent flow, where the energy is subsequently transferred to the ultimate heat sink by the Residual Heat Removal (RHR) system. As in previous pressure suppression containment designs, the maximum drywell pressure is limited to the wetwell pressure plus the vent submergence head and any vent system flow losses.

The operational modes of the PCC heat exchanger can best be described in terms of the pressure difference across the unit.

Figure A.3-3 illustrates several of a family of possible pressures along the flow path from the drywell to the suppression pool via the PCC heat exchanger. Note that on the drywell side, the pressure difference can vary only between that required to open the vacuum breaker and that required to open the main vent.

Reference LOCA Condition —

Curve 1 illustrates the SBWR post-LOCA condition with the PCC carrying the decay heat load. In this case, the drywell pressure is slightly greater than the PCC vent submergence pressure, but less than the LOCA vent submergence pressure. Thus water is forced out of the PCC vent line, clearing a gas venting path to the suppression pool. The flow is forced through the PCC heat exchanger by the drywell to wetwell pressure difference, and noncondensibles are vented into the suppression pool.

PCC Capacity Greater Than The Decay Heat —

Curves 2 and 3 of Figure A.3-3 illustrate a situation where most of the noncondensibles have been vented to the wetwell. These two curves illustrate two cases where the drywell is supplying nearly pure steam to the heat exchanger: Curve 3 has less noncondensibles than Curve 2. As the effects of noncondensibles degrading the heat transfer process are reduced, the heat exchanger can reject more energy than is supplied to the drywell by decay heat, and the drywell pressure is reduced. The reduced pressure is no longer capable of keeping the PCC vent open, so suppression pool water partially refills the PCC vent pipe. The flow into the PCC heat exchanger is no longer driven by the drywell-to-wetwell pressure difference, but by the lowered pressure in the heat exchanger tubes due to the condensation process. The limit of this type of operation is shown on Curve 4, where the drywell pressure has fallen to below the wetwell pressure by an amount equal to the vacuum breaker opening pressure. Here, the vacuum breaker opens, returning noncondensibles to the drywell to re-enter the PCCS. The capacity of the PCC to remove energy is temporarily degraded.

PCC Capacity Less Than The Decay Heat —

Finally, Curve 5 of Figure A.3-3 illustrates the other extreme of PCC operation. In this case, the PCC cannot remove sufficient heat to reject the decay heat, and the drywell pressure rises. Again, flow is forced through the PCC by the drywell-to-wetwell pressure difference. However, the magnitude of the PCC driving pressure difference is limited by the presence of the main LOCA vents. If the main LOCA vents clear, then mass and energy will flow to the suppression pool via the main vent system and limit the drywell pressure. This pressure difference also determines flow through the PCC heat exchanger.

In summary, there are two possible operating modes for the PCC heat exchanger: (1) a pressure drop driven mode, when the PCC vent is cleared of water, and flow is typically a mixture of steam and noncondensibles; and (2) a condensation pressure driven mode, when the PCC vent is partially filled with water, and the flow is nearly free of noncondensibles. These PCC operational modes are summarized below:

1. Pressure Drop Driven Mode
   - PCC capacity ≤ core decay heat
   - PCC flow is forced by the DW to WW pressure difference
   - PCC flow is a rich mixture of both steam and noncondensible gas

2. Condensation Pressure Driven Mode

- PCC capacity ≥ core decay heat
- PCC flow is induced by DW to PCC-Hx outlet pressure difference due to condensation
- PCC flow is rich in steam, but is lean in noncondensible gas

Steady-State Tests

The independent variables for the PANTHERS/PCC steady-state tests are steam flow rate, air flow rate, and PCC inlet pressure. The design basis of the Passive Containment Cooling System (3 heat exchangers) provides the ability to reject all SBWR decay heat at approximately one hour post-LOCA.

Figure A.3-4 compares the range of test conditions for PANTHERS/PCC with the air and steam flow conditions for the SBWR main steam line and GDCS line break scenarios after one hour into a LOCA. The test conditions clearly bound the possible SBWR range of air/steam and noncondensible flows. The third independent variable, PCC inlet pressure, is not indicated on this figure, but is shown for the various tests in Table A.3-2b. For this same time frame, the SBWR would be expected to have a PCC operating pressure near 300 kPa. Test Groups P3, P4, and P5 typically have data taken at five pressures, ranging from 200 to 500 kPa, with one pressure near the 300 kPa nominal value. All Test Group P6 data points are taken at a nominal PCC inlet pressure of 300 kPa, consistent with the P6 goal of increasing the data density near the post-LOCA SBWR operating conditions.

As noted in the previous discussion of operating modes, the pressure drop from the drywell (through the PCC heat exchanger to the PCC vent exit) cannot exceed a value equivalent to the difference between the main LOCA vent submergence and the PCCS vent submergence. The PCC pressure drop is one of the dependent variables measured during the testing. On the basis of this data, it is possible to establish the maximum flow rate through the PCC, independent of the time into a postulated LOCA scenario. This is the basis for using the PANTHERS/PCC data to qualify TRACG for application at times earlier than one hour post-LOCA.

Transient Tests

Transient tests are performed to assess two phenomena: the buildup of noncondensibles in the heat exchanger, and the reduction of PCC pool water level as the inventory is boiled away. In the noncondensible case, air and helium, representing heavier-than-steam and lighter-than-steam gases, are introduced at low volume flow rates; the flow rate is low enough such that the performance may be considered quasi-steady. The tests begin with pure steam condensation and noncondensibles are added until condensation is essentially stopped. Thus, the tests cover the entire potential range of PCC operation from the standpoint of noncondensible inventory in the condenser. In the water level tests, through a combination of normal boil-off and draining of the pool, the PCC pool level is lowered through a range that exceeds the SBWR inventory loss over a 72 hour period. Hence, both transient test types cover the entire applicable SBWR range.

Pressure Drop Driven and Condensation Pressure Driven Modes

As noted in the operational modes discussion, the PCC can perform in two modes: pressure drop driven and condensation pressure driven. Both of these conditions are simulated in the PANTHERS/PCC steady-state tests.

The pure steam tests, Test Conditions 37 through 49 (Test Groups P1 and P2) are all performed with the PCC vent closed. Since there is no vent flow through the heat exchanger, all the steam is condensed within the PCC and steam is drawn into the heat exchanger by the condensation process. These tests simulate the condensation pressure driven mode.

In the remaining air/steam mixture tests, Test Groups P3 through P6, the PCC vent is open, and both the inlet flow rate and vent tank pressure are controlled. These tests duplicate the pressure drop driven mode. In this case there is flow through the heat exchanger, with the flow rate determined by the difference in pressure between the inlet supply and the vent tank.

A.3.1.1.5 TRACG Analysis Plan

Table A.3-3 lists those PANTHERS/PCC tests that will be analyzed with TRACG.

Fifteen TRACG runs are included in this group, which is intended to demonstrate the capability of TRACG to predict the heat rejection rate of the PCC heat exchanger over a wide range of conditions. The focus will be on rated conditions, with the qualification points also established near the extremes of the SBWR range. Twelve of the qualification data points come from the steady-state performance test matrix (Test Groups P1 through P5), and the remaining three from the transient group (two from P7 and one from P8).

Figure A.3-5 illustrates the locations of the ten saturated condition steady-state TRACG qualification points within the overall PANTHERS/PCC steady-state test performance test matrix. The remaining two conditions are superheated, and cannot be shown on this figure.

Analysis results will be compared with test data as defined in Table A.3-3. For the steady-state saturated and superheated steam conditions, the assessment of adequacy will be made on the basis of *total heat rejection rate* and *PCC pressure drop*. For air/steam and helium/steam mixtures, the *degradation factor*, defined as the ratio of the heat rejection rate in the noncondensible case to that in the pure steam case, will be the figure of merit. The air/steam mixture data are taken at five different pressures. The degradation factor will be based on the air/steam mixture case having the absolute pressure nearest to the pure steam case:

Pure Steam Condensation — Analysis of Test Conditions 41 and 43 demonstrates TRACG capability to predict pure saturated steam condensation rates at and above rated conditions. Test Condition 49 addresses superheat in this state.

Air/Steam Mixtures — Analysis of Test Conditions 9, 15, 18, and 23 addresses the effects of noncondensible mass fraction at rated steam flow conditions, over the complete range of potential air fractions. Test Conditions 2 and 22 address the effects of air in the low steam flow range, but at the limits of air flows. Test Conditions 17 and 19 are in the intermediate range. Test Condition 35 addresses superheat effects.

Noncondensible Density — Analysis of Test Conditions 51 and 76 addresses the buildup of noncondensibles in the PCC tubes, and will be predicted on a transient basis. Test Condition 51 uses air and Test Condition 76 uses helium.

PCC Pool Level — Transient analysis of Test Condition 55 addresses the capability of TRACG to predict the effects of PCC pool water level.

A.3.1.2 PANTHERS/IC

A.3.1.2.1 Test Description

Overview

PANTHERS/IC (Isolation Condenser) testing is performed at Societa Informazioni Esperienze Termoidrauliche (SIET) in Piacenza, Italy. The tests are performed in the same facility used for the PANTHERS/PCC program, but using several pieces of different equipment, in order to better simulate the performance environment of the IC. For the IC testing, the facility consists of a prototype IC module, a steam supply vessel which simulates the SBWR reactor vessel, a vent volume, and associated piping sufficient to establish IC thermal-hydraulic performance. Both thermal-hydraulic and component demonstration tests are performed during these tests. This section covers the thermal-hydraulic portion of the testing; component performance tests are covered in Subsection A.3.2.2.

The IC being tested is one module of a full-scale, two-module vertical tube heat exchanger designed and built by Ansaldo. Only one module unit is being tested because of the much higher energy rejection rate of the IC relative to the PCC unit, and inherent limitations of facility and steam supply size. Figure A.3-6 gives an outline drawing of the heat exchanger assembly. Like the PCC unit, the IC is a prototype unit, built to prototype procedures and using prototype materials. Six modules (three heat exchanger units) of the type being tested are used in the SBWR. The IC is installed in a water pool having one half the appropriate volume for one SBWR IC assembly.

Instrumentation

Figure A.3-7 is a schematic of the PANTHERS/IC facility. The primary instrumentation specified is sufficient to ascertain heat exchanger thermal-hydraulic performance by performing mass and energy balances on the facility. Table A.3-4 defines the thermal-hydraulic measurements taken during the IC tests.

Like the PCC testing, all test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. References 51, 52 and 53 contain information on the IC instrumentation, as well as the PCC instrumentation.

Test Method

PANTHERS/IC testing procedures are specific to the type of test being performed. In general, however, the procedure for the steady-state tests will be as follows:

The steam vessel and IC heat exchanger will be purged of initial air in a manner similar to that done with the PCC heat exchanger. The IC pressure will be at the design pressure or a lower value, depending on whether the test is also being used as a structural demonstration cycle. The IC is placed in operation by opening the IC drain valve. Steam supply to the steam vessel is then regulated such that the vessel pressure stabilizes at the desired value. Data will be acquired for a period of approximately 15 minutes. At this point, the steam supply can be increased or decreased to gather data at a different operating pressure, or testing may be terminated. In all cases, flow into the IC will be natural circulation driven, as is the case for the SBWR.

Noncondensible gas effects tests begin similarly until the point where pressure is stabilized at the desired value. For this case, a mixture of air and helium is injected into the IC supply line at a very low flow rate. The ratio of air to helium in the injected flow will be 3.6:1, simulating the composition of radiolytic gases. Gas injection will continue until the IC inlet pressure increases to 7.653 mPag (1110 psig). (The noncondensible flow rate has not been determined, but will be specified in the Test Plan and Procedure, a program deliverable as noted in Table A.2-2.) The lower IC vent is then opened, and the IC vented until the pressure returns to the initial operating pressure, or stabilizes at an intermediate value. If the pressure returns to its initial value, the test is terminated. If the inlet pressure stabilizes, the IC top vent will be opened, and the performance monitored until venting is complete, and the inlet pressure returns to the initial value. The test is then terminated.

Water level tests also begin with the IC in stable operation at the desired initial inlet pressure. The IC pool water level is then reduced and the IC performance monitored. Water level will be reduced until the pool level is at mid height of the condenser tubes, or the IC inlet pressure reaches 8.618 mPag (1250 psig), whichever comes first. The pool water level will then be increased to normal and IC performance allowed to return to normal. The test is then terminated.

A.3.1.2.2 Test Objectives

The objectives of the PANTHERS/IC Test Program are:

1. Demonstrate that the prototype IC heat exchanger is capable of meeting its design requirements for heat rejection. *(Component Performance)*
2. Provide a sufficient data base to confirm the adequacy of TRACG to predict the quasi-steady heat rejection performance of a prototype IC heat exchanger, over a range of operating pressures that span and bound the SBWR range. *(Steady-State Separate Effects)*
3. Demonstrate the startup of the IC unit under accident conditions. *(Concept Demonstration)*
4. Demonstrate the capability of the ICC design to vent noncondensibles and to resume condensation following venting. *(Concept Demonstration)*

A.3.1.2.3 Test Matrix and Data Analysis

Steady-State Performance Tests

As for the PANTHERS/PCC tests, the majority of the IC tests are steady-state performance tests. Table A.3-5a provides the PANTHERS/IC Steady-State Performance Test Matrix. A total of ten test conditions are specified. Test Conditions 2 through 11 are identified as Test Group I1. These data will establish the IC heat rejection rate as a function of inlet pressure.

Transient Test Conditions

PANTHERS/IC transient tests will demonstrate startup of the IC heat exchanger for full-scale thermodynamic conditions. These tests are designed to demonstrate heat exchanger performance; they are not intended to be integral systems tests.

Table A.3-5b gives the PANTHERS/IC Transient Demonstration Test Matrix. Five Test Conditions are specified. Test Condition 1 (Test Group I2) is a set of two duplicate tests designed to demonstrate the startup and operation of the IC in a situation comparable to a reactor isolation and trip. Test Conditions 12 and 13 (Test Group I3) will have an air/helium mixture injected slowly after the steam vessel pressure has been reduced to the value specified as "inlet pressure" in Table A.3-5b. The IC will be vented when the inlet pressure reaches 7.653 mPag (1110 psig) or when the pressure peaks, if at a lower value. Re-establishment of condensation following venting will be recorded. Test Conditions 14 and 15 (Test Group I4) are repeats of Test Conditions 12 and 13, but with the water level in the IC pool allowed to drop, exposing the IC tubes. Both the IC pool level rate and the IC heat rejection rate will be monitored as a function of time.

- Test Group I2 will demonstrate startup of the IC under near prototype conditions, provide heat rejection data at a higher pressure than the data from Test Group I1, and demonstrate test repeatability. Test Conditions 12 and 13 will demonstrate restart of condensation in the IC following venting noncondensible. Test Conditions 14 and 15 will establish the degradation of heat rejection ability of the IC as the IC pool water level decreases.
- Test Groups I1 and I2 will be compared with design requirements to meet Test Objective 1.
- Test Groups I1, I2, and I4 provide a database for TRACG qualification and meets Test Objective 2.
- Test Group I2 demonstrates restart of the IC and meets Test Objective 3.

A.3.1.2.4 Justification of Test Conditions

Steady-State Tests

The independent variable for the PANTHERS/IC steady-state tests is the isolation condenser inlet pressure, which is equal to the steam vessel pressure. The isolation condenser is a natural circulation unit.

The IC inlet pressures to be tested shown in Table A.3-5a span the entire operating range of the SBWR. The SBWR range is bounded by the SRV setpoints at 7.920 mPag (1150 psig) and the vessel depressurized state. This is consistent with the test pressures.

Transient Tests

The transient test independent variables are IC inlet pressure, total noncondensible gas added, and IC pool water level. IC inlet pressures chosen are 0.48 mPag (70 psig) and 2.07 mPag (300 psig). These conditions were chosen because they represent typical non-LOCA operating conditions where an operator might have the IC in service. The ratio of air to helium in the injected gas was chosen to be representative of the oxygen to hydrogen ratio due to radiolytic decomposition of water in the SBWR core. While the injection rate has not be.n determined at this time, it will be chosen such that quasi-steady operation of the heat exchanger occurs.

For the water level tests, water levels as low as mid-height of he condenser tubes is specified. Bounding calculations based on decay heat rejection indicate that nc more than one-third of the tubes may be uncovered during the 72 hour post scram period. Consequently, the defined testing bounds the SBWR range of conditions.

A.3.1.2.5 TRACG Analysis Plan

Table A.3-6 lists those PANTHERS/IC tests that will be analyzed with TRACG. Six TRACG runs are included in this group, which is intended to demonstrate the capability of TRACG to predict the heat rejection rate of the IC heat exchanger over the range of reactor pressures where it will be expected to perform. Three of the six points come from the steady-state performance test matrix (Test Group I2), with the remaining three points coming from the transient data set.

Analysis will be compared with test data as defined in Table A.3-6. In all cases, the primary comparison will be on the *total heat rejection rate*. Additionally, for the transient cases, *IC inlet pressure* will be compared as a function of time:

Pure Steam Condensation — Analysis of Test Conditions 2, 6, and 11 demonstrates TRACG capability to predict pure steam IC condensation rates over the expected SBWR operating range (7.92 to 0.21 mPag) (1150 to 30 psig).

Noncondensible Buildup and Venting — Analysis of Test Conditions 12 and 13 demonstrates TRACG capability to predict the effect of noncondensible buildup in degradation of the overall heat transfer capability of the IC, including re-establishment of steam-only condensation following venting.

IC Pool Level Effects — Analysis of Test Conditions 15 demonstrates TRACG capability to predict the effect of pool level on the degradation of IC performance.

A.3.1.3 PANDA

A.3.1.3.1 Test Description

Overview

PANDA is a large-scale integrated SBWR containment experiment that will be performed by the Paul Scherrer Institut in Wuerenlingen, Switzerland. The test facility is an approximately 1/25 volumetric, full scale height simulation of the SBWR containment system. Pressure vessels representing the reactor pressure vessel, drywell, wetwell and wetwell air space, and GDCS pool are interconnected with appropriate piping in order to simulate a variety of containment transients. The facility is equipped with three scaled PCC heat exchangers and one isolation condenser unit, each with its own water pool. The PCC and IC units are both scaled by holding the heat transfer tubes at full size, but reduced in number from the prototype. The configuration of the IC and PCC units is illustrated on Figure A.3-8. The reactor pressure vessel volume is equipped with electrical heaters to simulate decay heat and thermal capacitance of the vessel and internals. The facility is capable of simulating SBWR accident scenarios starting approximately one hour into the LOCA.

Figures A.3-9 and A.3-10 show a schematic of the PANDA test facility and the arrangement of the PANDA test vessels, respectively. Two interconnected vessels are used for the drywell and wetwell volumes in order to simulate potential asymmetric effects.

In addition to its transient capabilities, PANDA also has temporary piping connections such that a PCC heat exchanger may be tested in a quasi-steady manner. In this case, a connection is made from the IC piping supply line to the inlet of PCC3. Steam can then flow directly from the RPV to PCC3, bypassing the drywells. PCC3 will vent to the wetwell and condensate will return to the GDCS tank, using the normal piping arrangement. The temporary supply piping arrangement is shown in Figure A.3-11.

After the IC and PCC units were fabricated, it was determined that a scaling compromise existed in the units. The fraction of heat removed through the headers was high compared to the prototype. This situation was caused by two factors: the increase in surface area to volume ratio, and a thinner metal thickness in the cylindrical section of the top and bottom headers in the test articles compared to the prototype. It was therefore determined that it would be best to add insulation to the headers to minimize this effect. Tne insulation design is shown in Figure A.3-12.

Instrumentation

The PANDA data acquisition system is capable of recording up to 720 channels with each channel recorded once every two seconds. For the PANDA tests, 598 channels have been assigned. The instrumentation is summarized in Table A.3-7, with approximate locations given in Figures A.3-13a through A.3-13d. Test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. Additional information may be found in the PANDA Test Specification, Reference 54, and in the Test Plan and Procedure, Reference 55.

For the steady-state PCC performance tests, only a subset of the PANDA instrumentation is required. This subset of the instrumentation is defined in Table A.3-8; locations are shown on Figure A.3-14.

Test Method

Steady-State Tests to demonstrate PCC performance will be the first tests performed in the PANDA facility. For these tests, the facility will be configured as described above and as shown schematically in Figure A.3-11.

The facility will be preconditioned for testing using the electrical heaters in the RPV for the heat source. The RPV will be filled with water to an appropriate level above the top of the heaters, and the heaters turned on. Once the water has been heated to saturation conditions, the RPV can be used to provide steam for heating of the other PANDA vessels. In addition, the hot water in the RPV will be used to heat water in the auxiliary water system. Then the steam, hot water, and/or air from the auxiliary water and air systems will be used to separately bring the GDCS tank, wetwell vessels, PCC3, and PCC3 pool to the desired pressures and temperatures.

Once the desired conditions are achieved in each vessel, the appropriate connecting lines will be opened, and the steam and air flow will be directed to PCC3. The power to the RPV heaters and the flow from the auxiliary air supply will be adjusted to obtain the desired steam and air flow rates, respectively. For the tests with no air flow, the PCC3 vent line will be closed and the condenser pressure will be allowed to come to the steady-state equilibrium value consistent with the specified steam flow rate.

After steady-state conditions have been achieved, the test will be initiated and the data will be recorded for a period of at least 15 minutes. The test will then be terminated.

Test procedures for the transient matrix tests have not yet been prepared. It is anticipated that facility pre-conditioning to establish the initial conditions for the transient tests will be similar to that described for the steady-state tests in the preceding paragraphs. Once the initial conditions for a given test have been established, all control (except for the decay of RPV power) will be terminated, and the PANDA containment will be allowed to function without operator intervention, mirroring the SSAR assumptions for the SBWR. Details will be submitted in the Test Plan and Procedure for these tests.

A.3.1.3.2 Test Objectives

The test objectives of the PANDA Test Program are:

1. Provide additional data to: (a) support the adequacy of TRACG to predict the quasi-steady heat rejection rate of a PCC heat exchanger, and (b) identify the effects of scale on PCC performance. *(Steady-State Separate Effects)*
2. Provide a sufficient database to confirm the capability of TRACG to predict SBWR containment system performance, including potential systems interaction effects. *(Integral Systems Tests)*
3. Demonstrate startup and long-term operation of a passive containment cooling system. *(Concept Demonstration)*

A.3.1.3.3 Test Matrix and Data Analysis

Steady-State Performance Tests

Two series of steady-state tests will be conducted using one of the PANDA PCC condensers. As noted in the test method section, the facility will be configured to inject known flow rates of saturated steam and air directly to the PCCS heat exchanger. The condenser inlet pressure will be maintained at approximately 300 kPa for all tests by controlling the wetwell pressure. The steam and air flow to the heat exchanger will be controlled and measured. In addition, the condenser drain flow will be measured.

Table A.3-9a shows the PANDA Steady-State PCC Performance test matrix. In the first series of tests, six test conditions (S1 through S6) are included. These tests will be performed with the headers of the PCC unit uninsulated. Pre-test analysis of the heat exchanger performance had already been submitted (Reference 56) when the decision was made to insulate the PANDA heat exchanger headers for matrix testing. It was decided to proceed with the tests as planned and add a second, abbreviated series of steady-state tests following the originally planned tests to establish any differences in heat exchanger performance between the insulated and uninsulated conditions. This second series of tests repeats three of the earlier test conditions. These tests are designated S7 through S9.

For both series of steady-state tests, the independent parameters are the steam and air mass flow rates. Conditions were chosen so that a direct comparison can be made to PANTHERS and GIRAFFE test points. Table A.3-9a identifies the test conditions in PANDA and the corresponding PANTHERS and GIRAFFE Test Conditions.

- Five Test Conditions (Test Conditions S1 through S5) are planned with various air flows and a constant steam flow of 0.195 kg/sec. In addition, one test will be performed with a pure steam flow equivalent to that expected to match the steam condensing capacity of the condenser (Test Condition S6). Test Conditions S7 through S9 duplicate earlier PANDA test points, but with the headers insulated.
- PANDA Test Conditions S1 through S9 provide a data base for TRACG qualification to meet the requirements of Test Objective 1(a).
- The results of PANDA Test Conditions S1 through S9 will be compared with the PANTHERS and GIRAFFE steady-state performance data as noted in Table A.3-9a to meet the requirements of Test Objective 1(b). The results of Test Conditions S7, S8, and S9 will be compared to Test Conditions S3, S5 and S6, respectively, to determine the effect of header insulation on PCC performance in PANDA and to baseline the PANDA PCC performance for the transient tests.

Transient Integral Systems Tests

A series of nine transient integral systems tests is planned for the PANDA facility to provide an integral systems database for PCC system performance with conditions representative of the long-term post-LOCA SBWR containment response. The philosophy used to determine the test matrix is to define a base case test representing SBWR performance under SSAR LOCA conditions, and then to perform perturbations around that base case to establish system effects and systems interaction effects. Two tests have been intentionally left undefined, so that the experience gained in the first seven tests may be utilized in their definition. Table A.3-9b summarizes the key characteristics of each test, and data use. It is planned to perform the tests in three groups of three tests each. The first group will consist of tests M3, M4, and M7, the second group tests M5, M6, and M8, and the final group the remaining tests M2, M9, and M10.

Test M3 will be the first matrix test performed and is identified as the *Base Case Test.* The initial conditions for Test M3 are summarized in Table A.3-10. These conditions were derived from the SBWR main steam break LOCA analysis at one hour after LOCA initiation. Additional information on the basis for this choice may be found in Subsection A.3.1.3.4.

The following provides the purpose and additional descriptive information on each PANDA transient test:

- *Test M1* was deleted and replaced with Test M10.
- *Test M2* is a perturbation to Test M3 with all of the break flow steam directed into drywell DW2. DW2 has two PCC condensers. This test maximizes the steam content of DW2 and the air content of DW1. It is the most asymmetric condition that can be established in PANDA. Test M2 results will be compared with Test M3 results to quantify asymmetric effects on PCCS containment performance.

- *Test M3* is the base case test, as defined in the previous paragraph.
- *Test M4* is a repeat of Test M3 to demonstrate transient system response repeatability.
- *Test M5* is a perturbation to Test M3, but with drywell spray actuation to initiate drywell depressurization and vacuum breaker cycling. Test initial conditions are the same as Test M3, but drywell spray will be initiated two hours after the start of the test. The spray will then be cycled on a one hour on, one hour off period, for the remainder of the test. The spray volumetric flow rate will be approximately 10 cubic meters per hour, with a spray temperature of approximately 40°C. Test M5 results will be compared with those of Tests M3 and M4 to evaluate the effects of active cooling systems on SBWR performance.
- *Test M6* is a perturbation to Test M3 with the IC operating in parallel with the three PCC condensers throughout the test period. This test will provide data showing the interaction between the PCC condensers and the IC, as well as the effect of the additional heat removal by the IC on containment and reactor system performance.
- *Test M7* will utilize the same nominal initial conditions as Test M3, but with the drywells and PCC units filled with air at the start of the transient. Additionally, this test will begin as early in the SBWR transient as is possible with the PANDA facility design. This test will provide data to determine the PCC condenser startup characteristics when initially blanketed with noncondensible gas.
- *Test M8* is a perturbation to Test M3, but with drywell-to-wetwell bypass leakage. The bypass leakage area will be set at ten times the allowable SBWR value as scaled to PANDA. This test will provide the effect of bypass leakage on containment performance.
- *Tests M9 and M10* will have test conditions defined later, utilizing the experience gained from the previous tests. These test will focus specifically on systems interactions.
- PANDA tests M2 through M10 provide a database for TRACG qualification that meets Test Objective 2.
- PANDA tests M2 through M10 address long-term operation of the PCCS. Tests M5 through M7, M9 and M10 address systems interaction and PCCS restart issues. These tests meet the requirements of Test Objective 3.

A.3.1.3.4 Justification of Test Conditions

Steady-State Tests

The conditions specified for PANDA Tests S1 through S9 were tabulated in Table A.3-9a. As noted in Subsection A.2.3, a philosophy of the SBWR test program is to test in multiple scales, wherever feasible. As noted in Table A.3-9a, every PANDA steady-state test shares test conditions with a condition from PANTHERS/PCC (See Subsection A.3.1.1) and GIRAFFE (See Subsection A.3.1.5). The specific conditions chosen duplicate PANTHERS/PCC Test Group P3 conditions at a steam flow near the mid-range for SBWR LOCA conditions and over a range of air flow fractions that bounds the SBWR range. Additionally, one pure-steam test was chosen at near the maximum for the SBWR range.

PANDA Tests S1-S6 are shown on the SBWR flow map in Figure A.3-15 This figure may be compared with Figure A.3-3 to see the similarity to the PANTHERS/PCC matrix. Tests S7 through S9 are repeats of Tests S3, S5, and S6, but with the PCC heat exchanger headers insulated. These conditions were chosen for repeat because they are at the maximum steam and air flow rates (Tests S9 and S8, respectively), and near the mid-range of the steam/air flow map (Test S7).

The choice of these test conditions was also chosen to facilitate comparison of TRACG predictions at different scales. Tests S1 through S6 all had TRACG pre-test analyses performed and submitted.

A.3.1.3.4.1 Transient Integral Systems Tests

Choice of the Base Case

The integral system response tests are specialized with the goal of investigating the highly ranked phenomena identified as Qualification Needs in Table 6.1-1. Since the number of tests is limited, the choice of conditions must be made to address the potential for systems interactions as well as individual system operations. Additionally, specific phenomena (e.g., drywell depressurization due to spray initiation, and PCC restart following noncondensible re-entry to the drywell) need to be addressed. Consequently, most integral systems test programs tend to be performed by definition of a *Base CaseTest,* around which perturbations are made to assess the effects of specific systems, systems interactions, and phenomena of interest. This testing philosophy was chosen for the PANDA program.

The choice of *Base Case Test* is central in this philosophy. For PANDA, the decision was made to use the SBWR main steam line break conditions at one hour post-LOCA initiation as the base case. This choice has both historical and technical reasons. Historically, GE pressure suppression containment and LOCA/ECCS testing has used conservative FSAR assumptions in definition of base cases. From a technical standpoint, this choice also is rational: SSAR conditions give conservative, yet realistic conditions from which to start an experiment. Since the process by which these conditions are predicated are mechanistic in nature, it is relatively straightforward to vary other conditions mechanistically to address the perturbations required. These arguments for using the SSAR conditions for the SBWR base case remain as valid today as they have been in the past. Since PANDA is primarily a containment response experiment, and the main steam line break is the limiting scenario for the SBWR, this scenario using SSAR assumptions was chosen for PANDA Test M3, the base case.

Determination of Base Case Initial Conditions

Initial conditions are chosen for the PANDA Test M3 base case on the basis of the predicted state of the RPV and containment at a one hour post-LOCA for a main steam line (MSL) break. The predictions are made using the SBWR TRACG integrated system containment model. This model incorporates a representation of the RPV and the associated systems (ADS, GDCS) which simulate a containment response starting from the beginning of the LOCA, i.e. the instant of the pipe break. The conditions at LOCA plus one hour are tabulated in Table A.3-11.

These conditions must then be synthesized to prescribe the initial thermodynamic state of the PANDA vessels representing the RPV, GDCS pools, drywell, wetwell, and PCCS pools. The process followed addresses the differences between the SBWR and test facility configurations, and averages multi-cell TRACG results into the single conditions possible to specify for the PANDA vessels. Facility limitations, such as dynamic load capability, must also be factored into the choice of conditions.

This process introduces three potential sources of discrepancy between the SBWR TRACG calculation and the test facility. The first results from averaging the conditions in the multi-cell SBWR model. For example, the SBWR model uses eighteen cells (Rings 3 and 4) to represent the drywell region above the RPV skirt. The PANDA vessels can be initialized at a single nominal drywell condition (total pressure, partial pressure of noncondensible, and temperature). The second potential source of discrepancy arises from the need to establish test facility conditions in which the vapor region in each vessel is in thermodynamic equilibrium with the vessel liquid. This is a practical consequence of the length of time it takes to pre-condition the facility and the absence of an independent means of heating the vapor regions of the vessels. The third potential source of discrepancy is introduced by the translation of instantaneous transient conditions from the SBWR model into initial steady-state conditions for the test facility.

The first two of these potential discrepancies may be resolved by comparing Tables A.3-10 and A.3-11. Typically, differences within the "SSAR" PANDA Test M3 conditions are small. For example, the eighteen cells in Rings 3 and 4 vary less than 1% in total pressure. The same is true for the RPV steam dome, and wetwell air space. Consequently, any departures from thermodynamic equilibrium are small. Likewise, since the variation in total pressures are small, the volume averaging used to determine the PANDA vessel pressures does not introduce a large error. In the drywell, the air partial pressures vary between 9 and 17 kPa, nearly a factor of 2, representing an expected variation in air distribution. Since it is impractical to produce a distribution of noncondensibles in the PANDA drywells, a volume weighted average is used.

This leaves the question of rate of change of the test conditions, the so-called "start-on-the-fly" approach. To address this issue, several key outputs from the TRACG SBWR simulation were investigated. Table A.3-12 presents the results of this investigation. The drywell pressure, wetwell pressure, wetwell air partial pressure, and the mid and upper drywell air partial pressures were chosen as key parameters, their time derivatives were calculated from the TRACG output. Comparing the derivatives with the PANDA initial conditions, all are seen to be at least four orders of magnitude less than the absolute values. Based on these results, the effect of starting the tests "on-the-fly" is judged to be negligible.

Other Tests

Once the base case is specified, system and phenomenological investigations may be performed by perturbations around this base case test.

The specific Qualification Needs on a test-by-test basis are listed in Table A.3-13.

A.3.1.3.5 TRACG Analysis Plan

Each of the six PANDA steady-state and the nine PANDA integral systems tests will have a TRACG analyses performed: post-test, or both pre- and post-test.

- ***Pure Steam Condensation*** — Analysis of Tests S1, S6, and S9 demonstrates TRACG's capability to predict pure saturated steam condensation rates at and above rated conditions.

- ***Air/Steam Mixtures*** — Analysis of Tests S2 through S5, S7 and S8 addresses the effects of noncondensible mass fraction in the PANDA PCC configuration.

- ***Drywell-Wetwell Noncondensible Distribution*** — Analysis of Tests M2, M3, and M5 through M10 addresses the effects of initial gas and vapor distribution within the containment system, including vacuum breaker flow, and demonstrate TRACG's capability to model integral systems performance.

- ***Systems Interactions*** — Analytical studies of systems interactions have identified vacuum breaker and IC operation as the most likely candidates for systems interaction effects. Analysis of Tests M5 and M6 address TRACG's capability to model systems interactions.

- ***Bypass Leakage*** — The TRACG analysis of Test M8 provides qualification of bypass leakage modeling.

A3.1.4 GIST

A3.1.4.1 Facility Description

The Gravity-Driven Integrated Systems Test (GIST) was performed by GE Nuclear Energy in San Jose, California, in 1988. Testing is complete, and results were reported in Reference 42. The GIST facility was a section-scaled simulation of the 1988 SBWR design configuration, with a 1:1 vertical scale and a 1:508 horizontal area scale of the RPV and containment volumes. Because of the 1:1 vertical scaling, the tests provided real-time response of the expected SBWR pressures and temperatures.

An integrated systems test was performed in order to include the effects of various plant conditions on GDCS initiation and performance. Figure A.3-16 provides a facility schematic, and Figure A.3-17 shows the major interconnecting lines. The GIST facility consisted of four pressure vessels: the RPV, upper drywell, lower drywell and the wetwell. The RPV included internal structures, an electrically heated core, and bypass and chimney regions.

Key interconnecting lines, such as drywell vents and depressurization lines with quenchers, were also included. The suppression pool/wetwell includes the water supply tank, a recirculation pump system used to heat and cool the pool water, and the air lines for pressurizing the wetwell air space.

The GIST facility was a simulation of the SBWR design as it existed in 1988. Several differences exist between the GIST configuration and the final SBWR design. These differences are listed and reconciled in Appendix B.

One hundred twenty test instruments were mounted on the vessels and piping in the GIST facility. These instruments were used to measure ADS initiation, drywell and pool temperatures, break flow rates, GDCS initiation and flow rates, and RPV conditions such as temperature, pressure and water level.

A3.1.4.2 Test Objectives

The test objectives for the GIST Test Program were:

1. Demonstrate the technical feasibility of the GDCS concept. *(Concept Demonstration)*
2. Provide a sufficient data base to confirm the adequacy of TRACG to predict GDCS flow initiation times, GDCS flow rates, and RPV water levels. *(Integrated Systems Test)*

A3.1.4.3 Test Matrix

The GIST Test Matrix is shown in Table A.3-14. Twenty-six test conditions were specified. These 26 individual tests were divided into four test types, three of them loss-of-coolant accidents:

- Bottom Drain Line Break (BDLB)
- Main Steam Line Break (MSLB)
- GDCS Line Break (GDLB)
- No-Break (NB)

A broad spectrum of test parameters was varied within each one of these test types. In each one of the four test categories, a base test was performed and then subsequent tests were run where only one parameter at a time was varied from that used in the base case. The GIST facility modeled SBWR plant behavior during the final stages of the RPV blowdown. The tests started with the vessel at 100 psig and continued until the GDCS flow initiated and flooded the RPV.

- *Series BDLB* (Bottom Drain Line Break) consisted of parametric variations around the base test case of a relatively small break below the core. Seven tests were run in this configuration.
- *Series MSLB* (Main Steam line Break) consisted of eight tests, six of which were parametric variations and two of which were duplicates to establish the repeatability of results.
- *Series GDLB* (GDCS Line Break) consisted of four tests. Variations in ADS configuration were the parameter in this series.
- *Series NB* (No-Break) consisted of seven tests. This series typically utilized conditions well removed from the SBWR 1988 design envelope. They form a data set at or outside the limits of SBWR, and are the most challenging for TRACG analysis. For example, this series included several tests where the wetwell initial pressure was atmospheric, and no air-purge occurred since there was no break. The major difference between the 1988 GIST and current SBWR configurations is the location of the GDCS pool. From the standpoint of GDCS injection, the GIST configuration is conservative

relative to the SBWR because the GDCS driving head is always slightly less in GIST than in the SBWR. In the case of zero wetwell pressure, the GDCS injection head is much less than in the SBWR. This makes GDCS injection in GIST more challenging.

- Analysis of GIST data as reported in Reference 42 has proven the technical feasibility of the GDCS concept and accomplishes Test Objective 1.
- The overall GIST database provides a sufficient basis for TRACG qualification and accomplishes Test Objective 2.

A.3.1.4.4 TRACG Analysis Plan

As part of the GIST program, five TRACG comparisons were previously performed. The objective of this effort was to confirm the capability of TRACG to accurately predict the GIST facility response to a variety of LOCA initiating events. The principal areas of interest were the effectiveness of the modeling of the GDCS and the modeling of the RPV and containment at low pressure conditions. The qualification consisted of post-test calculations with TRACG and comparison against GIST data. Comparisons were made for RPV pressure, RPV water level, core ΔP, GDCS flow rate, and GDCS initiation time. Good agreement was found between test and calculation; the results are reported in Reference 2.

GIST tests for which TRACG analysis were completed are identified in Table A.3-15. These tests represent the full spectrum of break types, a wide range of initial pressure vessel liquid inventory, variations of containment initial conditions, and several degrees of GDCS availability.

A.3.1.5 GIRAFFE

A.3.1.5.1 Test Description

Overview

GIRAFFE Isolation Condenser/Passive Containment Cooling testing was performed at the Toshiba Nuclear Engineering Laboratory in Kawasaki City, Japan. The results are reported in Reference 43. The test facility consisted of five major components which represent the SBWR primary containment and suppression chamber pools (S/C), the isolation condenser/passive containment cooling heat exchanger, and the connecting piping. Separate vessels represented the reactor pressure vessel, drywell, wetwell, GDCS and the IC/PCC pool, which houses the IC/PCC condenser unit. A schematic of the facility is shown in Figure A.3-18.

The IC/PCC condenser tested was a full-length, three-tube heat exchanger. The single unit could be utilized as either an IC or a PCC. Figure A.3-19 gives an outline drawing of the heat exchanger assembly. The IC/PCC was installed in a water pool composed of a makeup pool with a chimney and cavity arrangement in which the IC/PCC unit was set.

In July 1994, GE and the NRC Inspection Branch agreed that the existing GIRAFFE data, although completely adequate technically, did not meet the documentation standards required for an ANSI/ASME NQA-1 test. These GIRAFFE tests were performed as developmental tests. Consequently, less emphasis is being placed on this data to support SBWR Certification. Knowledge of SBWR performance gained from these tests remains germane to the certification effort, and no GIRAFFE documentation will be removed from the SBWR docket. Except for comparison of the GIRAFFE steady-state PCC performance data with that from PANTHERS and PANDA, no additional analysis of this data is planned. Information on this test program remains here for completeness.

A.3.1.5.2 Test Objectives

The objectives of the GIRAFFE Test Program are:

1. Provide a database to support primary data taken at other scales to confirm the capability of TRACG to predict the quasi-steady heat rejection rate of a PCC heat exchanger. *(Steady-State Separate Effects)*
2. Provide a database to support primary data taken at other scales to confirm the capability of TRACG to predict PCCS system performance. *(Integral Systems Tests)*

A.3.1.5.3 Test Matrix and Data Analysis

Steady-State Tests

The majority of the GIRAFFE data are steady-state performance data for the IC/PCC unit under PCC conditions. For these tests, the facility was placed in a condition where steam or nitrogen-steam mixtures were supplied to the IC/PCC; the condensed vapor and vented nitrogen were directed to volumes modeled to act as the reactor vessel and suppression chamber pool respectively. Condensate outlet flows from the IC/PCC were measured by measuring the RPV level increase, which, in turn, was used to determine heat removal rate by multiplying it by the latent heat of vaporization. The condensate was returned to the RPV, and the vented nitrogen was released to the S/C gas space. Once steady-state conditions were established, data were collected for a period of approximately 10 minutes. The time averaged data were reported and analyzed.

Table A.3-16 shows the GIRAFFE PCC Steady-State Performance Matrix used to provide data in support of the test objectives. Thirteen test conditions are included. These tests are identified in the test report as the Phase 1, Step 1 Tests, and comprise Test Group G1. These tests cover the SBWR range of steam and air mass flow rates, as has been previously discussed in the PANTHERS/PCC section. Data from Test Group G1 provide a support database for TRACG qualification and meet the requirements of Test Objective 1. Data from Test Group G2 will be compared to that from corresponding PANDA and PANTHERS tests to corroborate those results at a third scale.

A3.1.5.4 TRACG Analysis

A significant number of GIRAFFE TRACG comparisons have been performed as part of the qualification effort. The objective was to confirm the capability of TRACG to accurately predict PCC steady-state performance. Results are reported in Reference 2.

A.3.1.6 GIRAFFE/Helium

A.3.1.6.1 Test Description

Overview

The GIRAFFE/Helium tests are being performed by the Toshiba Corporation at their Nuclear Engineering Laboratory in Kawasaki City, Japan. The purpose of these tests is to demonstrate the operation of the passive containment cooling system (PCCS) in post-accident containment environments with the presence of a lighter-than-steam noncondensible gas as well as a heavier-than-steam noncondensible gas. These tests will demonstrate SBWR containment thermal-hydraulic performance, heat removal capability, and systems interactions. Also, they will provide additional data for the qualification of containment response predictions in the presence of lighter-than-steam noncondensible gases by the TRACG computer program.

The facility configuration is very similar to that used in the earlier GIRAFFE tests described in the previous section. The facility configuration is shown schematically in Figures A.3-20 and A.3-21. The primary facility changes from the earlier configuration include shortening the PCC tube length (to 1.8 meters) and modifying the piping orifices to yield flow resistances which more closely model the current SBWR values. Additionally, provision has been made for the continuous addition of helium to the drywell during a test. Details are provided in the GIRAFFE/Helium Test Specification (Reference 57).

The GIRAFFE/Helium tests are performed in accordance with Japanese Quality Assurance Standard JEAG-4101, 1990 (Reference 58). Review of this standard against the requirements of ANSI/ASME NQA-1 has shown that the essential elements of NQA-1 are met by this standard. Therefore, results from the GIRAFFE/Helium test program are appropriate for use as design basis data.

Instrumentation

Instrumentation utilized in the GIRAFFE/Helium test program is similar to that used in earlier GIRAFFE tests. Details are being finalized at the time of this TAPD revision, but test instrumentation will consist of approximately 120 thermocouple measurements, 20 pressure measurements, 40 differential pressure measurements, and 4 flow rate measurements. Test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. Detail of the instrumentation, including instrument lists, types, and ranges will be included in the Test Plan and Procedure.

Direct measurement of noncondensibles during the GIRAFFE/Helium test program will be performed by periodically taking samples of the process fluid at two points in the drywell and one point in the wetwell during all of the tests. Samples will be analyzed using gas chromatography. It is necessary to limit the total number of samples taken, so as not to affect the test results. The samples will be taken at the three locations specified, once per hour during the conduct of the test. This data will be used to validate indirect measurements of noncondensible concentration inferred from temperature measurements.

Method

GIRAFFE/Helium testing follows a methodology very similar to that used in PANDA. Once the initial conditions for a given test have been established, all control (except for the decay of RPV power and helium injection, if called for) will be terminated, and the GIRAFFE containment will be allowed to function without operator intervention, mirroring the SSAR assumptions for the SBWR. Details will be submitted in the Test Plan and Procedure for these tests.

A.3.1.6.2 Test Objectives

The test objectives of the GIRAFFE /Helium Test Program are:

1. Demonstrate the operation of a passive containment cooling system with the presence of a lighter-than-steam noncondensible gas, including demonstrating the process of purging noncondensibles from the PCC condenser. *(Concept Demonstration)*
2. Provide a database for computer codes used to predict SBWR containment system performance in the presence of a lighter-than-steam noncondensible gas, including potential systems interaction effects. *(Integral Systems Tests)*
3. Provide a tie-back test, which includes the appropriate Quality Assurance documentation to repeat a previous GIRAFFE test, thereby reinforcing the validity of the previous GIRAFFE testing.

A.3.1.6.3 Test Matrix and Data Analysis

Helium Test Series

The series of helium tests (designated as Test Group H) is performed to demonstrate the operation of the PCC system with the presence of a lighter-than-steam noncondensible gas. Four tests with lighter-than-steam, heavier-than-steam, and mixtures of heavier- and lighter-than-steam noncondensible gases are included. Table A.3-17 provides the test matrix which gives the initial drywell conditions and helium injection rate for each test. Each test will run for at least 8 hours, and demonstrate at least one purge/vent cycle of the PCC condenser.

The following provides the purpose and additional descriptive information for each GIRAFFE /Helium test:

- Test H1 is the base case with nominal initial conditions the same as in PANDA Test M3. Initial conditions are given in Table A.3-18
- Test H2 is a repeat of Test H1, but with helium replacing the total volume of nitrogen in the drywell and PCCS.
- Test H3 will have the same initial total drywell pressure as Tests H1 and H2, but with the initial noncondensible fraction consisting of a helium/nitrogen mixture.
- Test H4 will start with the same initial drywell conditions as Test H1, and will have constant helium injection to the drywell. The helium addition rate will be such that the helium is injected over a period of one hour. The helium injection will be terminated when the total mass of helium added is equal to the initial drywell helium mass for Test H3.

System response from the four tests will be compared to establish the effect of lighter-than-steam noncondensible, or a mixture of lighter-than-steam and heavier-than-steam noncondensibles, on the effectiveness of heat rejection by the IC/PCC heat exchanger.

GIRAFFE Tests H1 through H4 will demonstrate the startup and operation of the PCCS with the presence of a lighter-than-steam noncondensible gas. These tests meet the requirements of Test Objective 1.

GIRAFFE Tests H1 through H4 provide data for TRACG qualification to accomplish Test Objective 2.

"Tie-Back" Test Series

The "Tie-back" series of tests (designated as Test Group T) is performed to reinforce the validity of previous GIRAFFE testing that did not include sufficient documentation to qualify as design basis information. This series of two tests will be run in accordance with JEAG-4101 Quality Assurance Guidelines; in fact, one of these tests will be a repeat of an earlier GIRAFFE test. It is anticipated that the test results will match those of the earlier test, thus demonstrating its technical accuracy. Test T1, the test chosen for repeat, is a main steam line break test run following the tests documented in Reference 43. Test initial conditions are given in Table A.3-19.

Test T2 will have test conditions very similar to Test T1, but have initial drywell nitrogen content intermediate to Tests H1 and T1. Initial conditions for Test T2 are given in Table A.3-20

- Comparison of the results of Test T1 with the previous GIRAFFE main steam line break test results will meet the requirements of Test Objective 3.
- The combination of GIRAFFE/Helium Tests H1 through H4, T1, T2, and PANDA Tests M3/4, and M7 form a comprehensive data base for investigation of the startup of the PCC heat exchanger in the presence of noncondensibles, and meet the requirements of Test Objective 1.

A.3.1.6.4 Justification of Test Conditions

Choice of the Base Case

Test H1, defined as the Base Case for Test Groups H and T, utilizes the same initial conditions as PANDA Test M3 (see Table A.3-20). The justification for the M3 conditions given in Subsection A.3.1.3.4 also apply to GIRAFFE/Helium test H1.

The decision to use common initial conditions for the GIRAFFE/Helium and PANDA base cases is also advantageous from the test philosophy standpoint to test at different scales. Tests H1 and PANDA M3 may be compared directly to determine any effect of scale on the results.

Other Tests

The other tests specified as part of the GIRAFFE/helium program were defined in such a way as to investigate PCC startup and operation for a range of both lighter-than-steam and heavier-than-steam noncondensible conditions. Figure A.3-22 shows the initial conditions on an air/helium partial pressure map. Initial conditions for PANDA Tests M3 and M7 are also included on the figure. The figure clearly shows that PCC startup will be demonstrated over a very wide range of air/nitrogen to steam ratios, from nearly pure steam to pure air.

Bounding helium concentrations are difficult to specify, since post-LOCA hydrogen generation is not mechanistic, and therefore must be done in accordance with assumptions to accomodate federal regulations. Test H2 helium specification is unrelated to any scenario; all the nitrogen used in Test H1 is replaced with helium to obtain a one-to-one comparison of PCC system performance in the presence of lighter-than-steam and heavier-than-steam noncondensibles. Tests H3 and H4 are dependent upon the assumption of 100% metal water reaction to generate hydrogen over a one hour period. Due to the continuous purging of gases from the drywell to the wetwell during the time of hydrogen generation, the equilibrium concentration of hydrogen in the drywell is substantially less than would occur if all of the hydrogen were generated instantaneously. The [illegible]% helium partial pressure initial condition for Test H3 is based on this equilibrium value. Thus, Test H3 does not utilize a helium mass equivalent to hydrogen from a 100% metal water reaction in a SBWR, but about a fifth of that value.

A.3.1.6.5 TRACG Analysis Plans

All tests in the GIRAFFE/Helium H-series will have TRACG analysis performed on a blind post test basis. Although the tests will be performed prior to TRACG analysis, the analyst will have no knowledge of the test results while the analysis is being performed. Tests T1 and T2 will have TRACG analysis performed on a post-test basis.

A.3.1.7 GIRAFFE/SIT (Systems Interaction Test)

A.3.1.7.1 Test Description

Overview

The GIRAFFE/SIT (System Interaction Tests) will be performed by the Toshiba Corporation at their Nuclear Engineering Laboratory in Kawasaki City, Japan. Test data will be obtained for TRACG qualification during the late blowdown/early GDCS phase of liquid line breaks.

The facility configuration is discussed in Subsection A.3.1.6.1 and is shown schematically in Figure A.3-18, with the addition of a second heat exchanger so that both the PCC and IC can be in operation simultaneously. The configuration of the IC is similar to the PCC unit shown in Figure A.3-19.

The GIRAFFE/SIT tests will be performed in accordance with Japanese Quality Assurance Standard JEAG-4101, 1990 (Reference 58). Review of this standard against the requirements of ANSI/ASME NQA-1 has shown that the essential elements of NQA-1 are met by this standard. Therefore, results from the GIRAFFE/SIT test program are appropriate for use as design basis data.

Instrumentation

Instrumentation utilized in the GIRAFFE/SIT test program is similar to that used in earlier GIRAFFE tests. (See Subsection A.3.1.6.1.)

Method

GIRAFFE/SIT testing follows a methodology very similar to that used in PANDA and GIRAFFE/Helium. Once the initial conditions for a given test have been established, all control (except for the decay of RPV power and possibly the microheater power) will be terminated. The GIRAFFE RPV and containment will be allowed to function without operator intervention, mirroring the SSAR assumptions for the SBWR. Details will be identified in the Test Plan and Procedure for these tests.

A.3.1.7.2 Test Objectives

In the initial GE evaluation, no need for these tests was identified. However on page 16 of the TAPD Draft Safety Evaluation Report (DSER) the NRC staff notes, "While GE considers MSLBs to be the limiting accident in terms of containment performance, both GDCS line breaks and bottom drain line (BDL) breaks are more limiting in terms of reactor vessel response, especially minimum water level. The staff has, therefore concluded that additional integral systems tests are required as part of the design certification test program for the SBWR. The tests should be performed in an appropriately scaled facility that (a) represents the current design of the SBWR; (b) has the capability of simulating a range of design basis events, including GDCS line breaks and BDL breaks; and (c) has sufficient power and pressure capability to represent these events prior to the initiation of GDCS injection." The GIRAFFE facility meets these criteria.

Based on the above, the test objective of the GIRAFFE/SIT Test Program is:

Provide a data base to confirm the adequacy of TRACG to predict the SBWR ECCS performance during the late blowdown/early GDCS phase of a LOCA, with specific focus on potential systems interaction effects. (*Integral Systems Tests)*

A.3.1.7.3 Test Matrix and Data Analysis

A series of four transient systems tests is planned to provide an integral systems database for potential systems interaction effects in the late blowdown/early GDCS period. All four tests are liquid breaks: two GDCS line breaks and two bottom drain line breaks. Tests will be performed with and without the IC and PCC in operation, and two different single failures are considered. The test matrix defining the four tests is given in Table A.3-21. Preliminary initial conditions for the base case, Test GS1, are given in Table A.3-22.

The initial conditions for all tests approximate SBWR conditions 10 minutes post-LOCA, based on the breaks and equipment operations listed in Table A.3-21. All tests will run for approximately two hours. Containment related parameters will be taken from the appropriate SBWR TRACG LOCA case at the time RPV pressure is 1.034 mPa (150 psia).

The RPV collapsed water level at the start of the test will be determined by using the TRACG GIRAFFE model. Since GIRAFFE is not an exact "scale model" of the SBWR, it will not be practical to have the water/steam distribution in GIRAFFE be the same as in SBWR. For example, the GIRAFFE RPV lower plenum is shorter than the SBWR lower plenum. Additionally, the GIRAFFE RPV material is thinner, and begins the LOCA simulation at a lower temperature than the SBWR. As a result, a smaller amount of energy is transferred to the RPV lower plenum fluid in GIRAFFE. Methods to better simulate this energy addition are being investigated, and may effect the final definition of the initial RPV water level.

Additional details on the initial conditions for the other GIRAFFE/SIT tests will be included in the Test Plan and Procedure.

The following provides the purpose and additional information on each GIRAFFE/SIT test:

- Test GS1 is the base case test, a GDCS line break, with DPV failure as the single failure and neither the PCCS, nor the IC, in operation. This test has initial conditions similar to GIST Test C01A, and may be compared with GIST C01A to evaluate the effects of configuration distortions in GIST and potential GDCS containment system performance interactions.
- Test GS2 is a bottom drain line break, otherwise similar to Test GS1. Test GS2 results will be compared to those of Test GS1 to determine the effects of break location on minimum water level. Test GS2 will also be compared to GIST Test A01 in the same manner as Tests GS1 and C01A.
- Test GS3 is also a bottom drain line break with DPV failure, but for this test both the PCCS and IC will be functioning. Data from test GS3 will be compared directly with Test GS2 for identification of potential systems interactions associated with the IC and PCC.
- Test GS4 is a GDCS line break, with the single failure being a GDCS valve failure in one of the other GDCS injection lines. As in Test GS3, both the PCC and IC will be in operation. This condition is expected to provide the slowest rate of recovery of water level. Data from test GS4 can be compared to test GS1 to identify potential interactions with the IC and PCC even though the single failures are different.

GIRAFFE/SIT Tests GS1 though GS4 provide a data base for TRACG qualification that meets the GIRAFFE/SIT test objective.

A.3.1.7.4 Justification of Test Conditions

Choice of the Base Case Test

Test GS1, the base case test for this series, had conditions defined that resulted in the lowest predicted chimney water level, considering the various break locations, sizes, and single failure combinations. Additionally, the commonality of conditions between this case and that of GIST Test C01A allows a comparison between the GIST and GIRAFFE simulations. The differences between the GIST and GIRAFFE test configurations allow an assessment of the effect of containment on GDCS performance.

Other Tests

The other test cases were defined with the objective of identifying systems interactions, should they occur. Since the primary focus of this testing is GDCS performance, the RPV water level is the figure of merit in these investigations. TRACG predictions for several break locations, single failures, and IC/PCC operation combinations were performed. The additional tests were chosen based on these results, which are presented in Table A.3-23.

A.3.1.7.5 TRACG Analysis Plan

All four transient tests in the GIRAFFE/SIT series will have TRACG analysis performed on a blind post test basis. Although the tests will be performed prior to the TRACG analysis, the analyst will have no knowledge of the test results while the analysis is being performed.

Exceptions will be information needed to conduct the analysis such as actual initial conditions, decay power and microheater power during the test. The assessment of TRACG's adequacy will be based on the ability to predict chimney and downcomer water level.

A.3.1.8 Other Analyses Planned

The previous sections have discussed the major SBWR-unique test programs and defined the test conditions to be analyzed with TRACG.

This section will give a brief overview of these tests and the anticipated corresponding TRACG analyses.

A.3.1.8.1 1/6 Scale Boron Mixing Test

GE-NE has performed a set of boron mixing injection tests for BWR/5 and BWR/6 geometries. These tests were reported in Reference 28. The tests were performed in a 1/6 scale three-dimensional model of a 218 in. reactor pressure vessel, and used the High Pressure Core Spray (HPCS) spargers as the primary injection location of the simulated boron solution. Using scaled boron injection rates of either 400 or 86 gpm, with and without HPCS flow, the parametric effects on mixing were examined in the upper plenum and core bypass regions. Two alternate injection locations were also examined.

Standby Liquid Control injection locations are different in the SBWR from previous product lines, due primarily to the natural circulation recirculation feature of the SBWR. The SBWR utilizes direct injection into the core region through the shroud at 16 locations.

A series of TRACG predictions of the BWR/5-6 data is planned. Specific test cases to be analyzed have not yet been identified. Primary data comparisons will be made against data for the *mixing coefficient*, which is defined as the concentration of injected solution at the measured location divided by the concentration that would be present if the injected solution were uniformly mixed with the entire vessel inventory. Comparisons will be made at several locations.

A.3.1.8.2 CRIEPI Natural Circulation Thermal-hydraulic Test Facility

The CRIEPI test facility is a parallel channel test facility intended to study the stability characteristics of a natural circulation loop during startup conditions. The two parallel channels are 1.79m high and are equipped with heaters with a maximum power input of 64 kW each. At the channel exit, there is an adiabatic chimney which is 5.7m high. The loop has a separator, a condenser and a subcooler which are used to return the condensed steam to the downcomer. A preheater with a capacity of 150 kW controls the inlet temperature to the channels. Tests have been run at low pressure to simulate low pressure loop startup. Oscillations have been observed under some conditions and a stability map has been created for the test loop.

A.3.1.8.3 Dodewaard Plant Startup

The Dodewaard reactor is a natural circulation BWR with internal free surface steam separation. The reactor, with a maximum thermal power of 183 MWth, is connected to a turbogenerator capable of producing 60 MWe. Initial startup of the reactor was in 1969, and it has been operating continuously since that time. While relatively small in size, it is thermodynamically and neutronically similar to the SBWR. The SBWR startup procedures will be similar to those of Dodewaard.

On February 15 and 16, 1992, the reactor was started-up for its 23rd fuel cycle. During that startup, data were recorded to characterize the startup for potential TRACG analysis. Data were taken at discrete time intervals during the startup. Typically, the reactor was in a state of semi-equilibrium during the measurement. The results of the measurement show early establishment of recirculation flow during low power operation. No indication of any reactor instability, including geysering, was observed. Data are reported in References 15 and 45.

TRACG analysis of this startup is being performed.

A.3.1.8.4 Containment System Response — PSTF Mark III

In the early 1970s, GE-NE performed several series of tests at the Pressure Suppression Test Facility (PSTF) to support the Mark III containment design. The SBWR and Mark III containments share a similar horizontal vent system geometry.

The test series chosen for comparison is PSTF Series 5703, which was reported in Reference 20. Test Series 5703 utilized a full-scale, three horizontal vent system with geometry very similar to that used in the SBWR. Three comparisons will be performed to test data from Runs 5703-1, -2, and -3, for which simulated steam line break size was the primary variable.

A.3.1.8.5 Containment System Response — Mark II 4T

In the mid-1970s, GE-NE conducted a series of containment tests supporting the Mark II containment design in the 4T (Temporary Tall Test Tank) facility in San Jose, California.

Test Series 5101 is reported in Reference 38. These tests were a full-scale, single-vent simulation of Mark II (vertical vent pipe) performance. Normally, the drywell was heated to 150°C prior to test initiation to minimize steam condensation. One test, Run 33, used a unheated drywell. Very different response was seen due to steam condensation in the drywell. Additionally, Tests 34 and 35 were performed specifically to investigate the effect of a wetwell-to-drywell vacuum breaker. (In the Mark II containment, pressurization of the wetwell air space by pool swell causes a short term opening of the vacuum breaker.)

These three tests will be analyzed with TRACG.

A.3.1.8.6 Suppression Pool Stratification — PSTF

In the late 1970s, two series of experiments were performed in the PSTF specifically to investigate pool condensation and thermal stratification in the Mark III containment system. These data were initially reported in References 46 and 47, and extensively analyzed in Reference 48. More recently, these data were reviewed as one element of an effort to define an appropriate nodalization for the TRACG SBWR suppression pool, but specific comparisons to the data have not yet been performed.

The tests reported in Reference 46 utilized a full-scale single cell 9-degree segment of the Mark III vent system and suppression pool, while those reported in Reference 47 used a vent system and pool having the same full-scale height, but with flow areas and pool surface areas reduced by a factor of 3. Suppression pool temperatures were monitored by an array of thermocouples suspended throughout the pool. Initial pool temperatures and blowdown flow rates were measured.

TRACG will be used to analyze Test 5707 Run 1 and Test 5807 Run 29.

A.3.2 Component Demonstration Testing

A.3.2.1 PANTHERS/PCC

A.3.2.1.1 Test Description

Component testing of the prototype PCC heat exchanger is performed using the same hardware and test facility as described in Subsection A.3.1.1. The component demonstration tests are very similar in conduct to the thermal-hydraulic testing. The test article (PCC module "A") is instrumented with strain gages, accelerometers, and thermocouples. Structural instrumentation is shown on Table A.3-24. Data are collected during the thermal-hydraulic tests, as well as the structural performance tests described in this section.

A.3.2.1.2 Test Objectives

The test objective of the PANTHERS/PCC Component Demonstration Test is:

Confirm that the mechanical design of the PCC heat exchanger is adequate to assure its structural integrity over a lifetime that exceeds that required for application of this equipment to the SBWR.

A.3.2.1.3 Test Matrix and Data Analysis

The approach taken to address the test objective is to subject the equipment to a total number of pressure and temperature cycles well in excess of that expected over the anticipated SBWR lifetime. The test matrix is shown in Table A.3-25. The number of cycles was conservatively chosen as 10 LOCA cycles and 300 pressure test cycles. This represents five times the design requirement number of hypothetical LOCAs (2) and nearly 17 times the number of expected pneumatic test cycles in accordance with 10CFR50, Appendix J over the 60-year design life of the PCC (Reference 18). Note, credit is taken for the thermal cycles experienced during the PCC thermal-hydraulic testing in determination of this Component Demonstration Test Matrix.

Two types of tests are performed during the PANTHERS/PCC component demonstration test: simulated LOCA pressurizations and simulated pneumatic leak test pressurizations.

Simulated LOCA Pressurizations

Simulated LOCA cycles are performed by pressurizing the PCC units with steam to simulate both the temperature and pressure effects of a LOCA. The PCC pool is at ambient temperature at the beginning of a test, but is allowed to heat up to saturation as each cycle proceeds. Table A.3-26 gives the time history of the LOCA pressurizations. Each LOCA cycle lasts approximately 30 minutes. Ten cycles are performed.

Simulated Pneumatic Leak Test Pressurizations

Simulated pneumatic tests are performed by pressurizing the PCC heat exchanger with air to 758 kPag (110 psig). The PCC pool temperature is at ambient conditions during these pressurizations. The test pressure is held for 2 minutes for each cycle. A total of 300 cycles are performed. The test data will be analyzed by review of strains and acceleration data against component acceptance requirements, both in terms of magnitude and frequency content.

A.3.2.2 PANTHERS/IC

A.3.2.2.1 Test Description

Component testing of the prototype IC heat exchanger will be performed using the same hardware and test facility as described in Subsection A.3.2.1. The component demonstration tests will be very similar in conduct to the thermal-hydraulic testing. The test article (the IC condenser unit) is instrumented with strain gages, accelerometers, and thermocouples. Structural instrumentation is shown in Table A.3-27. Data will be collected during the thermal-hydraulic tests as well as the structural performance tests described in this section.

A.3.2.2.2 Test Objectives

The test objective of the PANTHERS/IC Component Demonstration Test is:

Confirm that the mechanical design of the IC heat exchanger is adequate to assure its structural integrity during the period of time between SBWR In-Service Inspections (ISI).

A.3.2.2.3 Test Matrix and Data Analysis

The approach taken to address the test objective is to include sufficient number of load cycles to reveal any thermal racheting where the elastically calculated stress levels exceed the ASME Code shakedown limits, so the measured deformations can be used to envelope the ASME alternative shakedown analysis approach. Specifically, it is planned to subject the IC to 20 load cycles with a large fraction of the cycles to include thermal transients which will be sufficient to meet the above criteria, as well as uncover unexpected vibrations or unacceptable crack indications at welds. Prototype non-destructive tests (NDT) will be performed before and after the cyclic testing. The test matrix is given as Table A.3-28. Note, credit may be taken for the thermal cycles experienced during the IC thermal-hydraulic testing in determination of this Component Demonstration Test

Matrix, provided that the cyclic structural test conditions are met during the thermal-hydraulic testing.

Simulated operational cycles will be performed by pressurizing the IC unit with steam, to simulate both the temperature and pressure effects of a LOCA. Tests will be performed at different pressures, and with varying pressurization rates and durations to simulate "normal" IC cycles, reactor heatup/cooldown cycles (without IC operation), and an ATWS event. The IC pool will be at ambient temperature at the beginning of a test, but will be allowed to heat up to saturation as each cycle proceeds. Cycles will last between 7 and 12 hours. Data will be recorded for durations of several minutes, periodically through each cycle. Cycle types are shown on Figure A.3-23.

The test data will be analyzed by review of strain and acceleration data against component acceptance requirements, both in terms of magnitude and frequency content. Evidence of crack initiation or growth will be obtained from comparison of the pre-test and post-test NDT.

A.3.2.3 Depressurization Valve (DPV)

A.3.2.3.1 Test Description

A Depressurization Valve (DPV) test program was performed to confirm the adequacy of a squib-actuated valve to provide a reliable means of rapidly depressurizing the reactor vessel. Performance tests were performed on the primer and propellant materials after exposure to the SBWR environmental conditions. Functional tests were performed on a full-scale prototype valve at the vendor's shop. The DPV was subjected to steam flow tests to measure the steam flow capacity and reaction loads. Finally, the DPV was subjected to accelerated environmental aging of the nonmetallic components, and dynamic testing. Results are reported in Reference 44.

A.3.2.3.2 Test Objectives

The test objectives of the DPV Test Program were:

1. Confirm that the DPV is a zero leakage valve, and that it opens on-demand with a momentary electrical signal, opens within the required response time, and remains open without an external power source.
2. Obtain data from flow testing to determine stresses in the DPV and confirm that the DPV saturated steam flow rate meets the minimum expected blowdown flow rate.
3. Obtain additional information on primer and propellant performance to provide evidence for later qualification testing.

A.3.2.3.3 Test Matrix and Data Analysis

Samples of the primer and propellant materials were subjected to irradiation, accelerated thermal aging, and LOCA steam aging. Firing tests were subsequently performed and the results confirmed that the pressure output versus response time met the performance requirements for the DPV.

Two full-scale prototype squib actuated DPVs were manufactured, assembled and tested by Pyronetics Devices, Inc., a subsidiary of OEA, Inc., of Denver Colorado. Firing tests were performed on a full-scale valve under both a high pressure (1500 psig) condition at the valve inlet and a low pressure (1 psig) condition at the valve inlet. A momentary electrical signal was supplied and it was confirmed that the valve opened within the required response time and remained open without an external power source. A thermal exposure heat transfer test was performed on the valve to assess the effects of ambient temperature and steam line temperature. It was confirmed that the booster surface temperature was acceptable when the valve was exposed to the SBWR environmental temperature conditions. A leakage test was performed for each valve metal diaphragm seal. Each seal was pressurized to 1650 psig and it was confirmed that there was zero leakage.

Flow and reaction load tests were performed on a full-scale valve at Wyle Laboratories of Huntsville, Alabama. The test facility was modified to incorporate a prototypical SBWR steam line section. The DPV was connected to this prototypical section and instrumented with pressure, temperature, and strain gages, accelerometers and displacement transducers. Four steam blowdown tests were performed. The test data confirmed that the DPV mass flow rate would be on the order of 2.4×10^6 lbm/hr at an operating pressure of 1100 psia.

Potential environmental qualification effects were investigated by addressing two elements. One element was the accelerated aging of those DPV components that contain non-metallic materials to ensure their reliability under adverse in-plant conditions. The second element was to subject a full-size prototype DPV to dynamically induced loads to simulate in-plant vibration. The booster assemblies with the non-metallic materials were subjected to accelerated aging conditions and then successfully fired, confirming that adequate pressure was delivered. The dynamic simulation was performed on a triaxial seismic table at Wyle Laboratories. The DPV was assembled using the aged components and then instrumented. The dynamic aging tests included resonance search, vibration exposure (slow sine wave sweep) and a series of triaxial multi-frequency random input motion tests. It was confirmed that when signaled to actuate, the DPV opened and remained open.

A.3.2.4 Vacuum Breaker Valve

A.3.2.4.1 Test Description

The vacuum breaker valve test program was designed to confirm that the vacuum breaker valve would provide a reliable leak tight boundary between the drywell and wetwell and prevent the pressure in the wetwell from exceeding that of the drywell by more than three pounds per square inch. Leak tightness is achieved by use of a nonmetallic main seal and a backup hard seat. The double seal design provides assurance that maximum leakage requirements will not be exceeded in the event that an obstruction should lodge on either seat. A full scale prototype valve was built and subjected to flow testing to verify lift pressure, flow capacity, and stability at low flow. The primary nonmetallic seal was radiation and thermally aged. Following thermal aging, the valve was dynamically aged and subjected to design basis accident conditions to confirm its leak tightness to steam. Finally, the fully aged valve was subjected to reliability testing to confirm that its intrinsic reliability was consistent with the assumptions of the SBWR PRA.

A.3.2.4.2 Test Objectives

The objectives of the vacuum breaker test program were to demonstrate that:

- The vacuum breaker flow capacity could be made equivalent to 1.04 square feet
- The vacuum breaker lift pressure was less than 0.5 psi.
- The disk was dynamically stable under low flow conditions.
- The hard seat equivalent flow area was less than 0.2 square centimeters.
- The main seal was air bubble tight as installed and has an equivalent leakage flow area of less than 0.02 square centimeters to steam in the fully degraded condition under design basis accident conditions.
- The dynamic loads which result in lift of the disk were acceptable.
- The opening and closing reliability are maintained after subjecting the fully aged valve to grit ingestion.

A.3.2.4.3 Test Matrix and Data Analysis

The vacuum breaker was air leak tested with a new seal and it was confirmed that the seal was bubble tight. The valve was then placed in the flow test facility and evaluated for lift pressure and low flow stability. The lift pressure and flow stability met requirements. The flow test demonstrated that the valve stroke was not sufficient to meet minimum flow requirements. Since the natural stability of the vaive eliminated the need for a disk damper, the stroke was increased to take credit for damper deletion. It was demonstrated that increasing the valve stroke results in achieving the required flow performance. A seal was then aged with radiation and placed in the valve for thermal aging. The valve leak test was then repeated and it was shown that the seal was air bubble tight.

The valve was then placed on a shake table for fragility testing to determine at what acceleration, lift occurred. The valve was then subjected to ten Safe Shutdown Earthquake acceleration time histories. Upon disassembly of the valve it was discovered that the ballast ring and the position sensor screws had come loose due to failure to engage existing lock washers. Screws had been ingested by the valve and hammered by the disk. Leak rate testing confirmed the main seal was undamaged and the hard seat still exceeded leak tightness requirements despite marring. The valve ruggedness and resistance to seal damage was demonstrated by this event.

The Design Basis Accident test demonstrated that the fully aged valve meets leak requirements at steam pressures and temperatures characteristic of a loss-of-coolant accident followed by water spray. The leak tightness of the valve was demonstrated by measuring the condensate from the steam that passed through the valve seals. During pressure peaks, water sprays and 80 hours of endurance testing, no measurable condensate leaked through the valve. The test demonstrated the inherent steam leak resistance of the valve.

The final test was the reliability testing, which subjected the fully-aged valve to grit ingestion to simulate possible environmental conditions that could affect bearing surfaces and seals during normal service. The valve was cycled three thousand times to demonstrate reliability at its required statistical failure rate of $3x10^{-4}$ per demand.

Table A.2-1 Thermal-Hydraulic Test Data Groups and Description

Facility	Data Group	Test Conditions	Description
PANTHERS/PCC	P1	7	PCC steady-state performance; saturated steam
PANTHERS/PCC	P2	6	PCC steady-state performance; superheated steam
PANTHERS/PCC	P3	4**	PCC steady-state performance; air/steam mixtures
PANTHERS/PCC	P4	7**	PCC steady-state performance; air/steam mixtures
PANTHERS/PCC	P5	2**	PCC steady-state performance; air/steam mixtures
PANTHERS/PCC	P6	14	PCC steady-state performance; air/steam mixtures
PANTHERS/PCC	P7	6	PCC performance; noncondensible buildup
PANTHERS/PCC	P8	3	PCC performance; water level effects
PANTHERS/IC	I1	10	IC steady-state performance; inlet pressure effects
PANTHERS/IC	I2	1*	IC start-up demonstration
PANTHERS/IC	I3	2	IC restart demonstration, noncondensible venting
PANTHERS/IC	I4	2	IC performance; water level effects
PANDA/PCC	S	9	PCC steady-state performance; steam and air/steam mixtures
PANDA	Phase 1	3	Containment performance
PANDA	Phase 2	3	Containment performance
PANDA	Phase 3	3	Containment performance
GIRAFFE	G1	13	PCC steady-state performance - steam and air/steam mixtures
GIRAFFE/Helium	G2	4	Containment performance - noncondensible density effects
GIRAFFE/Helium	G3	2	Containment performance - "Tie-back" test
GIRAFFE/SIT	G4	4	GDCS performance - integral systems tests
GIST	BDLB	7	GDCS performance - integrated system effects - bottom drain
GIST	MSLB	8	GDCS performance - integrated system effects - main steam
GIST	GDLB	4	GDCS performance - integrated system effects - GDCS breaks
GIST	NB	7	GDCS performance - integrated system effects - transients

* Test to be performed twice to demonstrate repeatability.

** Test to be performed five times at different absolute pressures.

Table A.2-2 SBWR Test Documentation Submittals

Test	Submittal Title	Document No.	Actual Submittal Date
PANDA			
	Test Specification	22A5587 Rev. 1	15 Feb 95
	As-Built Drawing Package		
	Instrumentation Drawing Package	MFN 044-95	27 Mar 95
	QA Implementation Procedures	PPCP-QA-01	16 Feb 95
	Pre-Test Analysis (S1-S6)	40315-NUC-94-7034	27 Feb 94
	Test Plan and Procedures (S1-S9)		
	Apparent Test Results (S1-S6)		
	Apparent Test Results (S7-S9)		
	Data Transmittal Report (S1-S9)		
	Test Plan and Procedure (M3, 4, 7)		
	Pre-test Analysis M3		
	Apparent Test Results (M3, 4, 7)		
	Data Transmittal Report (M3, 4, 7)		
	Test Plan and Procedure (M5, 6, 8)		
	Pre-Test Analysis M5		
	Apparent Test Results (M5, 6, 8)		
	Data Transmittal Report (M5, 6, 8)		
	Test Specification (update for M1&9)	22A5587 Rev. 2	
	Test Plan and Procedure (M1, 2, 9)		
	Pre-Test Analysis M2, M9		
	Apparent Test Results (M1, 2, 9)		
	Data Transmittal Report (M1, 2, 9)		
	PANDA Data Analysis Report		
GIRAFFE/Helium			
	Test Specification	25A5677	15 Feb 95
	As-Built Drawing Package		
	Instrumentation Drawing Package		
	QA Plan	TOGE110-T01	
	Test Plan and Procedures (T1, H1-H4, T2)	TOGE110-T07	
	Apparent Test Results (T1, H1, H2)		
	Data Transmittal Report (T1, H1, H2)		

Table A.2-2 SBWR Test Documentation Submittals (Continued)

Test	Submittal Title	Document No.	Actual Submittal Date
	Apparent Test Results (H3, H4)		
	Apparent Test Results (T2)		
	Data Transmittal Report (H3, H4, T2)		
GIRAFFE/SIT			
	Test Specification	25A5677 Rev. 1	
	Test Plan and Procedures (I1-I4)	TOGE110-T07 Rev. 1	
	Apparent Test Results		
	Data Transmittal Report		
	GIRAFFE Data Analysis Report		
PANTHERS/PCC			
	Test Specification	23A6999 Rev.3	15 Feb 95
	As-Built Drawing Package	many	30 Jun 94
	QA Plan	006-QQ-92	8 Sept 94
	Instrument Installation Spec.	00157ST92 Rev. 1	30 Jun 94
	Pre-Test Analyses	RAI 900.35	31 May 94
	Data Acquisition Spec.	0095RS91 Rev. 1	30 Jun 94
	Test Plan and Procedure	0098PP91 Rev. 1	16 Aug 94
	Process & Instrument Drawing	00209DD93 Rev. 4	12 Dec 94
	Data Transmittal Report		
	PANTHERS/PCC Data Analysis Report		
PANTHERS/IC			
	Test Specification	23A6999 Rev. 4	
	As-Built Drawing Package		
	Test Plan	0097PP91	
	Test Procedures		
	Data Acquisition Spec.	00306RS94	
	Process & Instrument Drawing	00210DD93	
	Apparent Test Results (Phase 1)		
	Pre-Test Analysis Package		
	Apparent Test Results (Phase 3)		
	Data Transmittal Report		
	PANTHERS/IC Data Analysis Report		

Table A.2-3 SBWR Analysis Documentation Submittals

<u>Pre-Test Predictions</u>
PANTHERS/PCC - Complete [59]
PANDA Steady-State (S Series) - Complete [56]
PANTHERS/IC
PANDA M2
PANDA M3
PANDA M5
PANDA M9
GIRAFFE/Helium (blind post-test)
GIRAFFE/SIT (blind post-test)
<u>Preliminary Validation Results</u>
PANDA Steady-State Tests
PANDA Transient Tests
PANTHERS/PCC
PANTHERS/IC
GIRAFFE/Helium
GIRAFFE/SIT

Table A.3-1 Required Thermal Hydraulic Measurements-PCC Test

Measurement	Units	Expected Range	Accuracy (2 Std. Dev.)	Frequency (samples per sec)
Pressures:				
Noncondensible gas inlet	kPa gage	0 - 760 (0 - 110)	2% *	0.1
Steam inlet	(psig)	0 - 760 (0 - 110)	2%	0.1
PCC inlet		30 - 690 (5 - 100)	2%	0.1
Condensate tank gas space		30 - 690 (5 - 100)	2%	0.1
PCC upper plenum		30 - 690 (5 - 100)	2%	0.1
Vent tank gas space		30 - 690 (5 - 100)	2%	0.1
Differential pressures:				
Condensate tank/vent tank	kPa (psi)	0 - 30 (0 - 5)	2%	0.1
Upper plenum/lower plenum		0 - 30 (0 - 5)	2%	1
Condensate tank/upper plenum		0 - 30 (0 - 5)	2%	1
Flow Rates:				
Steam inlet	kg/s (lb/s)	0 - 12 (0 - 25)	2%	0.1
Noncondensible inlet	kg/s (lb/s)	0 - 3 (0 - 5)	2%	0.1
Condensate	kg/s (lb/s)	0 - 12 (0 - 25)	2%	0.1
Vent line gas	kg/s (lb/s)	0 - 3 (0 - 5)	2%	0.1
Pool makeup	l/s (gpm)	0 - 13 (0 - 200)	2%	0.1
Temperatures:				
Steam inlet	°C (°F)	100 - 177 (212 - 350)	3 (5)	0.1
Noncondensible gas inlet		100 - 177 (212 - 350)	3 (5)	0.1
Upper plenum		100 - 171 (212 - 340)	3 (5)	0.1
PCC inlet		100 - 171 (212 - 340)	3 (5)	0.1
Lower plenum		10 - 171 (50 - 340)	3 (5)	0.1
Drain line		10 - 171 (50 - 340)	3 (5)	0.1
Drain tank		10 - 171 (50 - 340)	3 (5)	0.1
Vent line		10 - 171 (50 - 340)	3 (5)	0.1
Vent tank		10 - 171(50 - 340)	3 (5)	0.1
PCC pool (6 places)		10 - 100 (50 - 212)	3 (5)	0.1
Tube wall (inside & outside)		82 - 171 (180 - 340)	3 (5)	0.1
Pool makeup water		10 - 100 (50 - 212)	3 (5)	0.1
Water levels (collapsed):				
PCC pool	m (ft)	3.5 - 5.0 (11.5 - 16.4)	0.03 (0.1)	0.1
Drain tank		0 - 6.5 (0 - 21.2)	0.03 (0.1)	0.1
Drain line		0 - 6.0 (0 - 19.7)	0.03 (0.1)	0.1
Vent tank		0 - 6.5 (0 - 21.3)	0.03 (0.1)	0.1
Lower plenum,		0 - 3.0 (0 - 9.8)	0.03 (0.1)	0.1
Other (indirect):	MWth			
Heat rejection rate		0 - 15	0.3	0.02
System heat losses		0 - 0.5	0.05	0.02

* % means percent of full-scale

Table A.3-2a PANTHER/PCC Steady-State Performance Matrix - Steam Only Tests

Test Group Number	Test Condition Number	Steam Flow* [kg/s (lb/s)]	Air Flow* [kg/s (lb/s)]	Superheat† [°C(°F)]
P1	37	0.45(1.0)	0(0)	<10(18)
P1	38	1.4(3.0)	0(0)	<10(18)
P1	39	2.5(5.5)	0(0)	<10(18)
P1	40	3.6(8.0)	0(0)	<10(18)
P1	41	5.0(11.0)	0(0)	<10(18)
P1	42	5.7(12.5)	0(0)	<10(18)
P1	43	6.6(14.5)	0(0)	<10(18)
P2	44	1.4(3.0)	0(0)	15(27)*
P2	45	1.4(3.0)	0(0)	20(36)*
P2	46	1.4(3.0)	0(0)	30(54)*
P2	47	5.0(11.0)	0(0)	15(27)*
P2	48	5.0(11.0)	0(0)	20(36)*
P2	49	5.0(11.0)	0(0)	30(54)*

* Nominal Value

† Superheat conditions are relative to the steam partial pressure.

Table A.3-2b PANTHERS/PCC Steady-State Performance Matrix- Air-Steam Mixture Tests

Test Group Number	Test Condition Number	Steam Flow* [kg/s(lb/s)]	Air Flow* [kg/s(lb/s)]	Inlet Pressure* [kPa (psia)]	Superheat† [°C(°F)]
P3	9-1	5.0 (11.0)	0.076 (0.17)	296 (42.9)	<10 (18)
P3	9-2	5.0 (11.0)	0.076 (0.17)	330 (47.9)	<10 (18)
P3	9-3	5.0 (11.0)	0.076 (0.17)	385 (55.8)	<10 (18)
P3	9-4	5.0 (11.0)	0.076 (0.17)	549 (79.6)	<10 (18)
P3	9-5	5.0 (11.0)	0.076 (0.17)	703 (101.9)	<10 (18)
P3	9-6	5.0 (11.0)	0.076 (0.17)	782 (113.4)	<10 (18)
P3	15-1	5.0 (11.0)	0.16 (0.35)	300 (43.5)	<10 (18)
P3	15-2	5.0 (11.0)	0.16 (0.35)	329 (47.7)	<10 (18)
P3	15-3	5.0 (11.0)	0.16 (0.35)	441 (63.9)	<10 (18)
P3	15-4	5.0 (11.0)	0.16 (0.35)	500 (72.5)	<10 (18)
P3	15-5	5.0 (11.0)	0.16 (0.35)	648 (94.0)	<10 (18)
P3	15-6	5.0 (11.0)	0.16 (0.35)	790 (114.6)	<10 (18)
P3	18-1	5.0 (11.0)	0.41 (0.90)	284 (41.2)	<10 (18)
P3	18-2	5.0 (11.0)	0.41 (0.90)	300 (43.5)	<10 (18)
P3	18-3	5.0 (11.0)	0.41 (0.90)	328 (47.6)	<10 (18)
P3	18-4	5.0 (11.0)	0.41 (0.90)	467 (67.7)	<10 (18)
P3	18-5	5.0 (11.0)	0.41 (0.90)	599 (86.9)	<10 (18)
P3	18-6	5.0 (11.0)	0.41 (0.90)	641 (92.9)	<10 (18)
P3	23-1	5.0 (11.0)	0.86 (1.9)	296 (42.9)	<10 (18)
P3	23-2	5.0 (11.0)	0.86 (1.9)	329 (47.7)	<10 (18)
P3	23-3	5.0 (11.0)	0.86 (1.9)	437 (63.4)	<10 (18)
P3	23-4	5.0 (11.0)	0.86 (1.9)	505 (73.2)	<10 (18)
P3	23-5	5.0 (11.0)	0.86 (1.9)	584 (84.7)	<10 (18)
P4	2-1	1.4 (3.0)	0.014 (0.030)	179 (26.0)	<10 (18)
P4	2-2	1.4 (3.0)	0.014 (0.030)	201 (29.1)	<10 (18)
P4	2-3	1.4 (3.0)	0.014 (0.030)	299 (43.4)	<10 (18)
P4	13-1	2.5 (5.5)	0.16 (0.35)	244 (35.4)	<10 (18)
P4	13-2	2.5 (5.5)	0.16 (0.35)	296 (42.9)	<10 (18)
P4	13-3	2.5 (5.5)	0.16 (0.35)	383 (55.5)	<10 (18)

Table A.3-2b PANTHERS/PCC Steady-State Performance Matrix- Air-Steam Mixture Tests (Continued)

Test Group Number	Test Condition Number	Steam Flow* [kg/s(lb/s)]	Air Flow* [kg/s(lb/s)]	Inlet Pressure* [kPa (psia)]	Superheat† [°C(°F)]
P4	13-4	2.5 (5.5)	0.16 (0.35)	470 (68.2)	<10 (18)
P4	13-5	2.5 (5.5)	0.16 (0.35)	560 (81.2)	<10 (18)
P4	16-1	6.6 (14.5)	0.16 (0.35)	300 (43.5)	<10 (18)
P4	16-2	6.6 (14.5)	0.16 (0.35)	421 (61.0)	<10 (18)
P4	16-3	6.6 (14.5)	0.16 (0.35)	538 (78.0)	<10 (18)
P4	16-4	6.6 (14.5)	0.16 (0.35)	662 (96.0)	<10 (18)
P4	16-5	6.6 (14.5)	0.16 (0.35)	788 (114.3)	<10 (18)
P4	17-1	2.5 (5.5)	0.41 (0.90)	275 (39.9)	<10 (18)
P4	17-2	2.5 (5.5)	0.41 (0.90)	362 (52.5)	<10 (18)
P4	17-3	2.5 (5.5)	0.41 (0.90)	453 (65.7)	<10 (18)
P4	17-4	2.5 (5.5)	0.41 (0.90)	520 (75.4)	<10 (18)
P4	17-5	2.5 (5.5)	0.41 (0.90)	606 (87.9)	<10 (18)
P4	19-1	5.7 (12.5)	0.41 (0.90)	295 (42.8)	<10 (18)
P4	19-2	5.7 (12.5)	0.41 (0.90)	384 (55.7)	<10 (18)
P4	19-3	5.7 (12.5)	0.41 (0.90)	472 (68.4)	<10 (18)
P4	19-4	5.7 (12.5)	0.41 (0.90)	567 (82.2)	<10 (18)
P4	19-5	5.7 (12.5)	0.41 (0.90)	665 (96.4)	<10 (18)
P4	22-1	1.4 (3.0)	0.86 (1.9)	198 (28.7)	<10 (18)
P4	22-2	1.4 (3.0)	0.86 (1.9)	261 (37.8)	<10 (18)
P4	22-3	1.4 (3.0)	0.86 (1.9)	322 (46.7)	<10 (18)
P4	22-4	1.4 (3.0)	0.86 (1.9)	389 (56.4)	<10 (18)
P4	22-5	1.4 (3.0)	0.86 (1.9)	463 (67.1)	<10 (18)
P4	25-1	6.6 (14.5)	0.86 (1.9)	330 (47.9)	<10 (18)
P4	25-2	6.6 (14.5)	0.86 (1.9)	381 (55.2)	<10 (18)
P4	25-3	6.6 (14.5)	0.86 (1.9)	451 (65.4)	<10 (18)
P4	25-4	6.6 (14.5)	0.86 (1.9)	530 (76.9)	<10 (18)
P4	25-5	6.6 (14.5)	0.86 (1.9)	609 (88.3)	<10 (18)
P5	35-1	5.0 (11.0)	0.86 (1.9)	270 (39.2)	20 (36)*
P5	35-2	5.0 (11.0)	0.86 (1.9)	298 (43.2)	20 (36)*
P5	35-3	5.0 (11.0)	0.86 (1.9)	359 (52.1)	20 (36)*
P5	35-4	5.0 (11.0)	0.86 (1.9)	436 (63.2)	20 (36)*

Table A.3-2b PANTHERS/PCC Steady-State Performance Matrix- Air-Steam Mixture Tests (Continued)

Test Group Number	Test Condition Number	Steam Flow* [kg/s(lb/s)]	Air Flow* [kg/s(lb/s)]	Inlet Pressure* [kPa (psia)]	Superheat† [°C(°F)]
P5	35-5	5.0 (11.0)	0.86 (1.9)	499 (72.4)	20 (36)*
P5	35-6	5.0 (11.0)	0.86 (1.9)	587 (85.1)	20 (36)*
P5	36-1	5.0 (11.0)	0.86 (1.9)	263 (38.1)	20 (36)*
P5	36-2	5.0 (11.0)	0.86 (1.9)	341 (49.4)	20 (36)*
P5	36-3	5.0 (11.0)	0.86 (1.9)	422 (61.2)	20 (36)*
P5	36-4	5.0 (11.0)	0.86 (1.9)	507 (73.5)	20 (36)*
P5	36-5	5.0 (11.0)	0.86 (1.9)	558 (80.9)	20 (36)*
P6	1-1	0.45 (1.0)	0.014 (0.030)	300 (43.5)	<10 (18)
P6	3-1	2.5 (5.5)	0.027 (0.060)	300 (43.5)	<10 (18)
P6	4-1	3.6 (8.0)	0.027 (0.060)	300 (43.5)	<10 (18)
P6	5-1	5.0 (11.0)	0.027 (0.060)	301 (43.6)	<10 (18)
P6	6-1	5.7 (12.5)	0.027 (0.060)	304 (44.1)	<10 (18)
P6	7-1	6.6 (14.5)	0.027 (0.060)	301 (43.6)	<10 (18)
P6	8-1	1.4 (3.0)	0.076 (0.17)	300 (43.5)	<10 (18)
P6	10-1	5.7 (12.5)	0.076 (0.17)	308 (44.7)	<10 (18)
P6	11-1	6.6 (14.5)	0.076 (0.17)	308 (44.7)	<10 (18)
P6	12-1	0.45 (1.0)	0.16 (0.35)	300 (43.5)	<10 (18)
P6	14-1	3.6 (8.0)	0.16 (0.35)	303 (43.9)	<10 (18)
P6	20-1	5.0 (11.0)	0.59 (1.29)	303 (43.9)	<10 (18)
P6	21-1	6.6 (14.5)	0.59 (1.29)	353 (51.2)	<10 (18)
P6	24-1	5.7 (12.5)	0.86 (1.9)	352 (51.0)	<10 (18)

* Nominal Value

† Superheat referenced to steam partial pressure.

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Table A.3-2c PANTHERS/PCC Noncondensible - Buildup Matrix

Test Group Number	Test Condition Number	Steam Flow* [kg/s(lb/s)]	Helium Flow* [g/s]	Air Flow* [g/s]	Superheat† [°C (°F)]
P7	50	1.4 (3.0)	0 (0)	4.4	<10 (18)
P7	51	5.0 (11.0)	0 (0)	4.4	<10 (18)
P7	52	1.4 (3.0)	0 (0)	4.4	20 (36)*
P7	53	5.0 (11.0)	0 (0)	4.4	30 (54)*
P7	75	1.4 (3.0)	0.7	0 (0)	<10 (18)
P7	76	5.0 (11.0)	0.7	0 (0)	<10 (18)
P7	77	1.4 (3.0)	1.5	4.8	<10 (18)
P7	78	5.0 (11.0)	1.2	4.4	<10 (18)

* Nominal Value

† Superheat referenced to steam partial pressure.

Table A.3-2d PANTHERS/PCC Pool Water Level Effects - Test Matrix

Test Group Number	Test Condition Number	Steam Flow* [kg/s (lb/s)]	Air Flow* [kg/s (lb/s)]	Superheat† [°C (°F)]
P8	54	5.0 (11.0)	0 (0)	< 10 (18)
P8	55	5.0 (11.0)	0.14 (0.31)	< 10 (18)
P8	56	6.6 (14.5)	0.86 (1.9)	< 10 (18)

* Nominal Value

† Superheat referenced to steam partial pressure.

Table A.3-3 PANTHERS/PCC TRACG Qualification Points

Test Condition Number	Pre/Post Test Analysis	Data Comparison
41	Post	Heat Rejection Rate
		PCC Pressure Drop
43	Post	Heat Rejection Rate
9	Post	Heat Rejection Rate Degradation Factor
		PCC Pressure Drop
15	Pre/Post	Heat Rejection Rate Degradation Factor
		PCC Pressure Drop
18	Post	Heat Rejection Rate Degradation Factor
		PCC Pressure Drop
23	Pre/Post	Heat Rejection Rate Degradation Factor
		PCC Pressure Drop
2	Post	Heat Rejection Rate Degradation Factor
17	Post	Heat Rejection Rate Degradation Factor
19	Post	Heat Rejection Rate Degradation Factor
22	Post	Heat Rejection Rate Degradation Factor
35	Post	Heat Rejection Rate Degradation Factor
49	Post	Heat Rejection Rate
55	Post	Heat Rejection Rate
51	Post	Degradation Factor
76	Post	Degradation Factor

Table A.3-4 Required Thermal Hydraulic Measurements - IC Test

Measurement	Units	Expected Range	Accuracy (2 Std. Dev.)	Frequency (samples per sec)
Pressures:				
Steam vessel	mPa gage	0.4 - 10.34 (70 - 1500)	2% *	0.1
IC inlet	(psig)	0.4 - 10.34 (70 - 1500)	2%	0.1
IC upper plenum		0.4 - 10.34 (70 - 1500)	2%	0.1
Differential pressures:				
IC inlet/IC vent line	kPa (psi)	0 - 69 (0 - 10)	2%	0.1
IC inlet/IC drain line		0 - 69 (0 - 10)	2%	0.1
Upper plenum/lower plenum		0 - 69 (0 - 10)	2%	0.1
Elbow meter taps (4)		0 - ? (0 - ?)	2%	0.1
Flow rates:				
Steam inlet	kg/s (lb/s)	0 - 16 (0 - 35)	2%	0.1
Noncondensible inlet	kg/s (lb/s)	0 - 0.3 (0 - 0.5)	2%	0.1
IC pool makeup	l/s (gpm)	0 - 11.4 (0 - 180)	5%	0.1
Temperatures:				
IC inlet steam	°C (°F)	157 - 314 (315 - 598)	3 (5)	0.1
IC inlet pipe (6), (leak det.)		100 - 314 (212 - 598)	3 (5)	0.1
Drain line		10 - 314 (50 - 598)	3 (5)	0.1
Vent lines (2)		10 - 314 (50 - 598)	3 (5)	0.1
Steam vessel		65 - 314 (150 - 598)	3 (5)	0.1
IC pool (12 places)		10 - 104 (50 - 220)	3 (5)	0.1
Pool makeup water		10 - 104 (50 - 220)	3 (5)	0.1
Pool outlet temperature		10 - 104 (50 - 220)	3 (5)	0.1
Tubes (3 @ 5 axial locations)		10 - 314 (50 - 598)	3 (5)	0.1
Water levels (collapsed):				
IC pool	m (ft)	3.5 - 5.5 (11.5 - 18.0)	0.03 (0.1)	0.1
Simulated RPV		later (later)	0.03 (0.1)	0.1
Drain line		later (later)	0.03 (0.1)	0.1
Vent lines (2)		later (later)	0.03 (0.1)	0.1
Other (indirect):				
IC heat rejection rate	MWth	0 - 20	0.1	0.02
System heat loss	MWth	0 - 1	0.1	0.02

* % means percent of full-scale

Table A.3-5a PANTHERS/IC Steady-State Performance - Test Matrix

Test Condition Number	Test Group No.	Inlet Pressure [mPag (psig)]
2	I1	7.920 (1150)
3	I1	7.240 (1050)
4	I1	6.21 (900)
5	I1	5.52 (800)
6	I1	4.83 (700)
7	I1	4.14 (600)
8	I1	2.76 (400)
9	I1	1.38 (200)
10	I1	0.69 (100)
11	I1	0.21 (30)

Table A.3-5b PANTHERS/IC Transient Demonstration - Test Matrix

Test Condition Number	No. of Cycles	Test Group Number	Initial Pressure, P1 [mPag(psig)]	Inlet Pressure [mPag (psig)]	Initial Pool Temp. [(°C °F)]
1	2	I2	9.480 (1375)	8.618 (1250)	<21 (70)
12	1	I3	0.48 (70)	0.48 (70)	saturation
13	1	I3	2.08 (300)	2.08 (300)	saturation
14	1	I4	0.48 (70)	0.48 (70)	saturation
15	1	I4	2.08 (300)	2.08 (300)	saturation

Table A.3-6 PANTHERS/IC TRACG Analysis Cases

Test Condition Number	Pre/Post Test	Data Comparison
2	Post	Heat Rejection Rate
6	Pre/Post	Heat Rejection Rate
11	Post	Heat Rejection Rate
12	Post	Heat Rejection Rate Inlet Pressure
13	Pre/Post	Heat Rejection Rate Inlet Pressure
15	Post	Heat Rejection Rate

Table A.3-7 PANDA Instrumentation Summary (Including auxiliary systems instrumentation)

Measurement Type	Instrument Type	Number	Total
Temperature	Chromel-alumel-thermocouples	442	
	Pt100-Resistance thermometers	21	
	Thermistors (TC ref. temp.)	30	493
Pressure	Rosemount Model 3051CA transducer	15	
	Rosemount Model 2088A transducer	3	
	Rosemount Model 1144A transducer	3	21
Pressure difference	Rosemount Model 3051CD transducer	14	
	Rosemount Model 1151DP transducer	13	27
Level	Rosemount Model 3051CD transducer	7	
	Rosemount Model 1151DP transducer	11	18
Gas concentration	Oxygen partial pressure probe	2	
	Other (to be determined)	6	8
Flow rate	Vortex flow meter	11	
	Ultrasonic flow meter	3	
	Hot film flowmeter	1	15
Fluid phase detector	Conductivity probe	9	9
Electrical power	Wattmeter	6	
	Electronic totalizer	1	7
Total			598

Table A.3-8 Instrumentation Required for Test S1 to S9

Ident. Code	Description
MV.I1F	Steam flow to PCC3
MM.BOG	Air flow to PCC3
MV.P3C	PCC3 condensate flow (PCC3 to GDCS)
MV.GRT	PCC3 condensate flow (GDCS to RPV)
MV. P3V	PCC3 Vent flow to WW2
ML.U3	PCC3 pool level
ML.RP.1	RPV level
MP.I1F	PCC3 upper header pressure
MP.RP.1	RPV pressure
MP.P3V	PCC3 vent line pressure
MTG.P2F.1	Air/steam temperature in steady-state supply line
MTG.P3F.1	Steam temperature in steady-state supply line
MTL.P3C.1	PCC3 condensate temperature at GDCS inlet
MTL.GRT.1	PCC3 condensate temperature in GDCS drain line
MTG.P3V.1	Gas temperature in PCC3 vent line
MTL.P3C.2	PCC3 condensate temperature in PCC3 outlet
MTL.GRT.2	PCC3 condensate temperature at RPV inlet
MTG.P3V.2	Gas temperature in PCC3 vent line outlet at PCC3
many	PCC3 temperature*

* It is required that 30% of the pool temperature sensors and 50% of the tube wall and fluid sensors be available. The available pool sensors must include at least one of the three lowest elevations. The available tube wall and fluid sensors must include at least 40% of the probes above and below the horizontal mid-plane of the tube bundle. Within these constraints, the test engineer has the responsibility and authority to judge whether or not sufficient PCC3 temperature sensors are operable to initiate tests.

Table A.3-9a PANDA Steady-State PCC Performance Test Matrix

PANDA Test No.	Header Insulation	Steam Flow (kg/s)	Air Flow (kg/s)	PANTHERS Test Condition No.	GIRAFFE Phase 1, Step 1 Test No.
S1	no	0.195	0	41	2
S2	no	0.195	0.003	9	4
S3	no	0.195	0.006	15	6
S4	no	0.195	0.016	18	8
S5	no	0.195	0.034*	23	10
S6	no	0.26	0	43	3
S7	yes	0.195	0.006	15	6
S8	yes	0.195	0.034*	23	10
S9	yes	0.26	0	43	3

* It may not be possible for the PANDA air supply to deliver this flow rate. If this flow rate cannot be reached, then the test will be run at the maximum air flow rate that can be reached.

Table A.3-9b PANDA System Test Matrix Summary

PANDA Test No.	Break Type	No. of PCC	Drywell Spray	IC Operation	Bypass Leakage Area	Initial Conditions	Comments
M1	Test Deleted						
M2	MSL -0% to DW1 -100% to DW2	1 in DW1 2 in DW2	no	no	0	SSAR	Asymmetric steam flow to DW1 and 2
M3	MSL 50% to each DW	1 in DW1 2 in DW2	no	no	0	SSAR	Base Case Same as GIRAFFE/HE test H1
M4	Same as M3	1 in DW1 2 in DW2	no	no	0	SSAR	Repeatability
M5	Same as M3	1 in DW1 2 in DW2	Yes	no	0	SSAR	Drywell spray to initiate vacuum breaker operation
M6	Same as M3	1 in DW1 2 in DW2	no	Yes	0	SSAR	IC operation
M7	Same as M3	1 in DW1 2 in DW2	no	no	0	DW and PCC initially air-filled	Bounding case for PCC start-up
M8	Same as M3	1 in DW1 2 in DW2	no	no	10 times allowable	SSAR	DW to WW bypass leakage
M9	Conditions to be defined later						
M10	Conditions to be defined later						

Table A.3-10 Initial Conditions for PANDA Test M3

	RPV	Drywell	Wetwell	GDCS	PCC Pools
Total Pressure (kPa)	295	294	285	294	101
Air/Nit. Pressure (kPa)	0	13	240	274	n/a
Vapor Temperature (K)	406	405	352	333	n/a
Liquid Temperature (K)	406	405	352	333	373
Collapsed WL (m) (1)	11.2(2)	(2)	3.8	10.7	23.2

Notes:

(1) Water levels are specified relative to the top of the PANDA heater bundle.

(2) The nominal DW condition is no water. However, a small amount of spill from the RPV to the DW at the start of the test is acceptable.

Table A.3-11 SBWR Containment Conditions at 3600 sec for Main Steam Line Break LOCA

Level/Ring	Ring 1(R=2.5m)	Ring 2(R=3.0m)	Ring 3(R=4.26m)	Ring 4(R=7.8m)	Ring 5(R=13.75m)	Ring 6(R=15.75m)
Level 9 Z=24.812m	vol=29.688 m^3 dw head p=293.85 kPa pa=101.95 kPa alp=1.000 Tsat=391.43 K Tv=394.56 K Tl=n/a	vol=13.063 m^3 dw head p=293.85 kPa pa=50.79 kPa alp=1.000 Tsat=398.80 K Tv=409.63 K Tl=n/a	vol=43.452 m^3 upper dw p=293.85 kPa pa=9.34 kPa alp=1.000 Tsat = 403.93 K Tv=439.97 K Tl=n/a	vol=202.792 m^3 upper dw p=293.85 kPa pa=9.66 kPa alp=1.000 Tsat = 403.9 K Tv=440.18 K Tl=n/a	not used	not used
Level 8 Z=23.30m	vol=85.726 m^3 rpv-stm dome p=294.71 kPa pa=3.91 kPa alp=0.908 (level) Tsat=404.66 K Tv=424.40 K Tl=404.65 K	vol=37.720 m^3 rpv-stm dome p=294.70 kPa pa=3.10 kPa alp=0.909 (level) Tsat=404.75 K Tv=419.81 K Tl=405.26 K	vol=196.606 m^3 upper dw p=293.89 kPa pa=9.23 kPa alp=1.000 Tsat=403.95 K Tv=439.92 K Tl=n/a	vol=917.570 m^3 upper dw p=293.89 kPa pa=8.91 kPa alp=1.000 Tsat=403.99 K Tv=437.24 K Tl=n/a	vol=232.509 m^3 gdcs pl-up cell p=293.90 kPa pa=48.12 kPa alp=0.785 (level) Tsat=399.16 K Tv=417.02 K Tl=335.24 K	vol=465.047 m^3 gdcs pl-up cell p=293.90 kPa pa=49.37 kPa alp=0.786 (level) Tsat=398.95 K Tv=414.92 K Tl=333.41 K
Level 7 Z=19.60m	vol=20.263 m^3 rpv-sep region p=321.90 kPa pa=0.00 kPa alp=0.302 (mixture) Tsat=408.90 K Tv=408.09 K Tl=408.22 K	vol=50.178 m^3 rpv-up dwncmr p=321.64 kPa pa=0.00 kPa alp=0.017 (mixture) Tsat=408.06 K Tv=408.06 K Tl=407.88 K	vol=127.528 m^3 upper dw p=293.94 kPa pa=10.75kPa alp=1.000 Tsat=403.78 K Tv=433.94 K Tl = n/a	vol=595.180 m^3 upper dw p=293.94 kPa pa=9.60 kPa alp=1.000 Tsat=403.91 K Tv=437.75 K Tl=n/a	vol=150.817 m^3 gdcs pl-low cell p=313.08 kPa pa=n/a alp=0.000 Tsat=330.66 K Tv=n/a Tl=330.66 K	vol=301.652 m^3 gdcs pl-low cell p=313.04 kPa pa=n/a alp=0.000 Tsat=333.41 K Tv=n/a Tl=333.41 K
Level 6 Z=17.20m	vol=6.597 m^3 rpv-stnd pipes p=332.29 kPa pa=n/a alp=0.000 Tsat=408.89K Tv=n/a Tl=409.15 K	vol=12.493 m^3 rpv-dwncmr p=335.26 kPa pa=n/a alp=0.000 Tsat=407.77 K Tv=n/a Tl=407.77 K	vol=31.882 m^3 upper dw p=293.97 kPa pa=16.54 kPa alp=1.000 Tsat=403.10 K Tv=412.54 K Tl=n/a	vol=148.795 m^3 upper dw p=293.97 kPa pa=15.70 kPa alp=1.000 Tsat=403.20 K Tv=415.65 K Tl=n/a	not used	not used
Level 5 Z=16.60m	vol=93.030 m^3 rpv-chimney p=355.91 kPa pa=n/a alp=0.000 Tsat=409.15 K Tv=n/a Tl=409.15 K	vol=40.436 m^3 rpv-dwnmcr p=358.90 kPa pa=n/a alp=0.000 Tsat=407.36 K Tv=n/a Tl=407.34 K	vol=244.429 m^3 drywell p=294.01 kPa pa=16.56 kPa alp=1.000 Tsat=40-6.10 K Tv=412.42 K Tl=n/a	vol=1140.763 m^3 drywell p=294.01 kPa pa=16.51 kPa alp=1.000 Tsat=403.11 K Tv =412.73 K Tl=n/a	vol=1852.985 m^3 ww-vap space p=284.79 kPa pa=255.43 kPa alp=1.000 Tsat=344.83 K Tv=346.52 K Tl=n/a	vol=852.628 m^3 ww-vap space p=284.79 kPa pa=262.35 kPa alp=1.000 Tsat=339.62 K Tv=342.06 K Tl=n/a
Level 4 Z=12.00m	vol=40.448 m^3 rpv-chimney p=385.89 kPa pa=n/a alp=0.000 Tsat=409.13 K Tv=n/a Tl=409.13 K	vol=17.624 m^3 rpv-dwncmr p=388.93 kPa pa=n/a alp=0.000 Tsat=407.14 K Tv=n/a Tl=407.13 K	vol=106.273 m^3 drywell p=294.07 kPa pa=16.55 kPa alp=1.000 Tsat=403.11 K Tv=412.40 K Tl=n/a	vol=495.984 m^3 drywell p=294.07 kPa pa=16.53 kPa alp=1.000 Tsat=403.11 K Tv=412.65 K Tl=n/a	vol=805.646 m^3 ww-vap space p=284.88 kPa pa=244.15 kPa alp=0.839 (level) Tsat=351.62 K Tv=351.39 K Tl=352.31 K	vol=370.708 m^3 ww-vap space p=284.88 kPa pa=248.41 kPa alp=0.838 (level) Tsat=349.27 K Tv=348.85 K Tl=351.97 K

Table A.3-11 SBWR Containment Conditions at 3600 sec for Main Steam Line Break LOCA

Level/Ring	Ring 1(R=2.5m)	Ring 2(R=3.0m)	Ring 3(R=4.26m)	Ring 4(R=7.8m)	Ring 5(R=13.75m)	Ring 6(R=15.75m)
Level 3 Z=10.00m	vol=44.493 m^3 rpv-chimney p=404.97 kPa pa=n/a alp=0.000 Tsat=408.50 K Tv=n/a Tl-408.49 K	vol=19.387 m^3 rpv-dwncmr p=408.05 kPa pa=n/a alp=0.000 Tsat=406.82 K Tv=n/a Tl-=406.82 K	vol=33.868 m^3 drywell p=294.10 kPa pa=18.32 kPa alp=1.000 Tsat=403.11 K Tv=412.40 K Tl=n/a	vol=158.157 m^3 drywell p=294.11 kPa pa=18.30 kPa alp=1.000 Tsat=403.12 K Tv=412.53 K Tl=n/a	vol=261.432 m^3 ww-sup pool p=298.42 kPa pa=n/a alp=0.000 Tsat=353.10 K Tv=n/a Tl=352.32 K	vol=120.295 m^3 ww-sup pool p=298.44 kPa pa=n/a alp=0.000 Tsat=352.07 K Tv=n/a Tl-352.06 K
Level 2 Z=7.80m	vol=18.555 m^3 rpv-core bypass p=429.45 kPa pa=n/a alp=0.000 Tsat=405.00 K Tv=n/a Tl=404.99 K	vol=18.233 m^3 rpv-dwncmr p=432.37 kPa pa=n/a alp=0.000 Tsat=406.36 K Tv=n/a Tl=406.36 K	vol=142.033 m^3 drywell p=294.15 kPa pa=16.56 kPa alp=1.000 Tsat=403.12 K Tv=412.29 K Tl=n/a	vol=26.000 m^3 drywell p=294.15 kPa pa=16.55 kPa alp=1.000 Tsat=403.12 K Tv=412.37 K Tl=n/a	vol=549.430 m^3 ww-sup pool p=323.73 kPa pa=n/a alp=0.000 Tsat=352.08 K Tv=n/a Tl=352.03 K	vol=252.814 m^3 ww-sup pool p=323.73 kPa pa=n/a alp=0.000 Tsat=351.99 K Tv=n/a Tl=351.99 k
Level 1 Z=4.65m	vol=63.912 m^3 rpv-lwr plenum p=466.17 kPa pa=n/a alp=0.000 Tsat=403.87 K Tv=n/a Tl=403.87 K	vol=18.078 m^3 rpv-lwr plenum p=466.18 kPa pa=n/a alp=0.000 Tsat=405.86 K Tv=n/a Tl=405.85 K	vol=31.804 m^3 lower dw p=294.22 kPa pa=16.54 kPa alp=1.000 Tsat=403.13 K Tv=412.30 K Tl=n/a	vol=148.433 m^3 lower dw p=294.22 kPa pa=16.53 kPa alp=1.000 Tsat=403.13 K Tv=412.3 K Tl=n/a	not used	not used
TEE35	vol=187.2 m^3(cell 3) lwr dw, Z=-3.4m p=294.29 kPa pa=12.87 kPa alp=1.000 Tsat=403.6 K Tv=410.8 K Tl-n/a	vol=233.0 m^3 (cell 2) lwr dw, Z=-6.8m p=294.35 kPa pa=9.86 kPa alp=1.000 Tsat=403.9 K Tv=408.4 K Tl=n/a	vol=213.0 m^3 (cell 1) lwr dw, Z=-10.0m p=305.89 kPa pa=10.94 kPa alp=0.189 (level) Tsat=405.1 K Tv=405.1 K Tl=405.1 K			

Table A.3-12 Time Derivatives of Key PANDA Initial Conditions

Parameter	PANDA Initial Condition (kPa)	Time Derivative SBWR@ t=3600 (pa/sec)	Ratio (1/sec)
Drywell Pressure	294	0.899	3.06×10^{-6}
Wetwell Pressure	285	0.899	3.15×10^{-6}
Wetwell Air Partial Pressure	240	1.29	5.38×10^{-6}
Mid DW Air Partial Pressure	13	0.11	8.38×10^{-6}
Upper DW Air Pressure	13	-1.726	-1.33×10^{-4}

Table A.3-13 PANDA TRACG Analysis Cases

Test Number	Pre/Post Test	Data Comparison
S1	Pre/Post	Heat Rejection Rate
S2	Pre/Post	Heat Rejection Rate Degradation Factor
S3	Pre/Post	Heat Rejection Rate Degradation Factor
S4	Pre/Post	Heat Rejection Rate Degradation Factor
S5	Pre/Post	Heat Rejection Rate Degradation Factor
S6	Pre/Post	Heat Rejection Rate
S7	Post	Heat Rejection Rate Degradation Factor
S8	Post	Heat Rejection Rate Degradation Factor
S9	Post	Heat Rejection Rate
M1	DELETED	
M2	Pre/Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. Suppression Pool Temp. PCC Flows
M3	Pre/Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. PCC Flows Suppression Pool Temp.
M4	Same as M3	

Table A.3-13 PANDA TRACG Analysis Cases

Test Number	Pre/Post Test	Data Comparison
M5	Pre/Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. Suppression Pool Temp. PCC Flows Vacuum Breaker Flows
M6	Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. Suppression Pool Temp. PCC Flows IC Flow
M7	Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. Suppression Pool Temp. PCC Flows
M8	Post	Drywell Pressure Drywell air distribution Wetwell Pressure Wetwell air distribution Drywell Temp. Wetwell Temp. PCC Flows Suppression Pool Temp. Leakage Flow
M9	Pre/Post	To Be Determined
M10	Post	To Be Determined

Table A.3-14 GIST Test Matrix Initial Conditions (RPV at 100 psig)

Test (1)	No. of GDCS Lines	RPV Level (in)(2)	Scram Time (sec)(3)	Decay Heat (kW)	LDW Level (in)	UDW Press. (psig)	S/P Level (ft)	S/P Temp. (°F)	WW Press. (psig)
BDLB Tests:									
A01 Base Case	3	347	369	89	4	13.0	67.2	105	6.5
A02 Low S/P Water Level	3	347	369	89	4	13.0	59.2	105.	6.5
A03 Maximum GDCS Flow	4	347	369	89	4	13.0	67.2	105	6.5
A04 Low RPV Water Level	3	327	369	89	4	13.0	67.2	105	6.5
A05 CRD Level	3	347	369	89	4	13.0	67.2	105	6.5
A06 Minimum GDCS Flow	1	347	369	89	4	13.0	67.2	105	6.5
A07 No Low Press DPVs	3	347	369	89	4	13.0	67.2	105	6.5
MSLB Tests:									
B01 Base Case	3	340	212	99	6	14.5	67.2	110	7.0
B02 Low PRV Water Level	3	320	212	99	6	14.5	67.2	110	7.0
B03 Low S/P Water Level	3	340	212	99	6	14.5	67.2	110	7.0
B04 First Repeat Test	3	340	212	99	6	14.5	67.2	110	7.0
B06 Last Repeat Test	3	340	212	99	6	14.5	67.2	110	7.0
B07 Low-Low RPV WL	3	300	212	99	6	14.5	67.2	110	7.0
B08 Accumulator Makeup	3	300	212	99	6	14.5	67.2	110	7.0
B09 Accumulator Makeup	3	286	212	99	6	14.5	67.2	110	7.0
GDLB Tests:									
C01A Base Case	2	347	373	88	5	11.5	67.2	105	7.0
C02 Max HP DPV Area	2	347	373	88	5	11.5	67.2	105	7.0
C03 Min HP DPV Area	2	347	373	88	5	11.5	67.2	105	7.0

Table A.3-14 GIST Test Matrix Initial Conditions (RPV at 100 psig) (Continued)

Test [1]	No. of GDCS Lines	RPV Level (in)[2]	Scram Time (sec)[3]	Decay Heat (kW)	LDW Level (in)	UDW Press. (psig)	S/P Level (ft)	S/P Temp. (°F)	WW Press. (psig)
C04 High LP DPV Setpt.	2	347	373	88	5	11.5	67.2	105	7.0
NB Tests:									
D01A Base Case	3	347	865	74	0	0.0	67.2	107	0.0
D02 Maximum GDCS Flow	4	347	865	74	0	0.0	67.2	107	0.0
D03A App. K Decay Heat	3	347	865	94	0	0.0	67.2	107	0.0
D04 Pressurized WW	3	347	865	74	0	14.7	67.2	107	14.7
D05 High Pool Temp.	3	347	865	74	0	0.0	67.2	157	0.0
D06 Low GDCS Injection	4	347	865	74	0	0.0	67.2	107	0.0
D07 No Power	3	347	—	0	0	0	67.2	107	0.0

Notes:

(1) Suffix "A" in Test Number signifies a repeat test.

(2) Collapsed water level relative to bottom of RPV.

(3) Time since reactor scram in SBW. Used to determine decay heat.

Table A.3-15 GIST Runs With Existing TRACG Analysis

Run	Type
B01	MSLB, Base Case
B07	MSLB, Low Initial RPV Level
C01A	GDLB, Base Case
A07	BDLB, No Low Pressure DPVs
D03A	NB, Zero Containment Pressure

Table A.3-16 GIRAFFE Test Matrix (Phase 1 Step-1)

Test No.	Test Group	Steam Flow Rate (kg/s)	Nitrogen Partial Pressure (fraction of total press.)	Pressure (kPa)
1	G1	0.02	0	300
2	G1	0.03	0	300
3	G1	0.04	0	300
4	G1	0.03	0.01	300
5	G1	0.02	0.02	300
6	G1	0.03	0.02	300
7	G1	0.04	0.02	300
8	G1	0.03	0.05	300
9	G1	0.02	0.10	300
10	G1	0.03	0.10	300
11	G1	0.04	0.10	300
12	G1	0.03	0.02	300
13	G145	0.03	0.02	300

Table A.3-17 GIRAFFE/Helium Integral Systems Test Matrix

GIRAFFE Test No.	Helium Injection Rate (Kg/sec)	Drywell Initial Partial Pressures (kPa) (±2kPa)		
		Nitrogen	Steam	Helium
H1	0	13	281	0
H2	0	0	281	13
H3	0	13	214	67
H4	0.00027	13	281	0

Table A.3-18 GIRAFFE/Helium Base Case (H1) Initial Conditions

Parameter	Value	Tolerance
RPV Pressure (kPa)	295	±6 kPa
Initial Heater Power (kW)	41+heat loss compensation	±1 kW
RPV Collapsed Water Level (m)*	13.2	±0.150 m
Drywell Pressure (kPa)	294	±4 kPa
Wetwell Pressure (kPa)	285	±4 kPa
Wetwell Nitrogen Pressure (kPa)	240	±4 kPa
GDCS Gas Space Pressure (kPa)	294	±4 kPa
GDCS Nitrogen Pressure (kPa)	274	±4 kPa
Suppression Pool Temperature (K)	352	±2 K
PCC Pool Temperature (K)	373	±2 K
GDCS Pool Temperature (K)	333	±2 K
GDCS Pool Level * (m)	†	
Suppression Pool Level* (m)	3.2	±0.075 m
PCC Pool Collapsed Water Level* (m)	23.2	±0.075 m
PCC Vent Line Submergence (m)	0.90	±0.075 m

* Referenced to the Top of Active Fuel (TAF)

† GDCS pool level should be positioned in hydrostatic equilibrium with the RPV level (including an appropriate adjustment for temperature difference).

Table A.3-19 GIRAFFE/Helium "Tie-Back" Initial Conditions

Parameter	Value	Tolerance
RPV Pressure (kPa)	189	±6 kPa
RPV Collapsed Water Level (m)*	9.1	±0.150 m
Initial Heater Power (kW)	96	±1 kW
Drywell Total Pressure (kPa)	188	±4 kPa
Drywell Nitrogen Partial Pressure(kPa)	53	±4 kPa
Drywell Steam Partial Pressure (kPa)	135	±4 kPa
Wetwell Pressure (kPa)	174	±4 kPa
Wetwell Nitrogen Pressure (kPa)	164	±4 kPa
GDCS Pool Gas Space Total Pressure (kPa)	188	±4 kPa
GDCS Pool Gas Space Nitrogen Partial Pressure(kPa)	151	±4 kPa
Suppression Pool Temperature (K)	326	±2 K
PCC Pool Temperature (K)	373	±2 K
GDCS Pool Temperature (K)	350	±2 K
GDCS Pool Level* (m)	14.1	±0.075 m
Suppression Pool Level *(m)	3.5	±0.075 m
PCC Pool Collapsed Water Level * (m)	23.2	±0.075 m
PCC Vent Line Submergence (m)	0.90	±0.075 m

* Referenced to the TAF.

Table A.3-20 GIRAFFE/Helium Test T2 Initial Conditions

Parameter	Value	Tolerance
RPV Pressure (kPa)	267	±6 kPa
Initial Heater Power (kW)	41+heat loss compensation	±1 Kw
RPV Collapsed Water Level (m)*	13.2	±0.150 m
Drywell Pressure (kPa)	266	±4 kPa
Drywell Nitrogen Pressure (kPa)	38	±4 kPa
Wetwell Pressure (kPa)	266	±4 kPa
Wetwell Nitrogen Pressure (kPa)	212	±4 kPa
GDCS Gas Space Pressure (kPa)	266	±4 kPa
GDCS Nitrogen Pressure (kPa)	246	±4 kPa
Suppression Pool Temperature (K)	352	±2 K
PCC Pool Temperature (K)	373	±2 K
GDCS Pool Temperature (K)	333	±2 K
GDCS Pool Level (m)	†	
Suppression Pool Level* (m)	3.2	±0.075 m
PCC Collapsed Water Level *(m)	23.2	±0.075 m
PCC Vent Line Submergence (m)	0.90	±0.075 m

* Referenced to the TAF.

† GDCS pool level should be positioned in hydrostatic equilibrium with the RPV level (including an appropriate adjustment for temperature difference.

Table A.3-21 GIRAFFE/SIT Test Matrix

Test	Break	Single Failure	IC/PCCS on?
GS1	GDL	DPV	No
GS2	BDL	DPV	No
GS3	BDL	DPV	Yes
GS4	GDL	GDCS	Yes

GDL = Gravity Drain Line
BDL = Bottom Drain Line
DPV = Depressurization Valve
GDCS = GDCS Injection Valve

Table A.3-22 Test GS1 Initial Conditions

Parameter	Value	Tolerance
RPV Pressure (kPa)	1034	20
RPV Collapsed Water Level (m)*	-2.34	0.15
Initial Heater Power (kW)	68+ heat loss compensation	1
Drywell Total Pressure (kPa)	289	4
Drywell Steam Partial Pressure (kPa)	178	4
Wetwell Total Pressure (kPa)	254	4
Suppression Pool Temperature (K)	333	2
PCC Pool Temperature (K)	373	2
GDCS Pool Temperature (K)	319	2
GDCS Pool Level (m)*	16.2	0.075

* Referenced to Top of Active Fuel (TAF).

Table A.3-23 Basis for GIRAFFE/SIT Test Conditions

Objective	Option			Test ID
	Break	Failure	IC/PCC Operation	
Worst Break/Single Failure Combination	GDL	DPV	No	GS1
Benefit of IC/PCC	BDL and BDL	DPV DPV	Yes No	GS3 GS2
Slow Water Level Recovery	GDL	GDCS	Yes	GS4
Fast Water Level Recovery	BDL BDL	DPV DPV	No Yes	GS2 GS3
Case representing a different break than worst break	BDL	DPV	No	GS2
Case showing GDCS void quenching and break flow depressurizing drywell	BDL GDL	DPV DPV	No No	GS2 GS1

Table A.3-24 PANTHERS PCC Structural Instrumentation

Measurement/Location	No. of Positions	Quantity at each Position	Total Measure-ments	Direction(s)
Acceleration:				
Steam distributor	1	3	3	X, Y, Z
Mid-length of tube	5	2	10	X, Y
Upper header cover	1	3	3	X, Y, Z
Displacement:				
Inlet/header junction	1	2	2	X, Z
Steam distributor	1	1	1	Z
Lower header support	2	1	2	Y
Total Strain:				axial
Inlet elbow	1	2	2	Z
Inlet/header junction	1	2	2	Z
Upper header/tube junction	5	1 or 2	7	Z
Tube/lower header junction	3	1	3	X, Y
Lower header	2	2	4	Z, X
Lower header cover	1	2	2	X, Z
Upper header	2	4	8	X, Z
Upper header cover	1	4	4	Y
Upper header cover bolts	3	1 or 2	5	Y
Lower header cover bolts	3	1 or 2	5	X, Z
Drain/lower header junction	1	2	2	Z
Lower header supports	1	2	2	
Permanent strain:				
Inlet/header junction	1	1	1	Z
Upper header/tube junction	3	1	3	Z
Lower header/drain junction	1	2	2	Z
Temperature:				
Steam line	2	1	1	1
Temperature:				
inlet/header junction	1	1	1	
Upper header/tube junction	3	1	3	
Tube/lower header junction	3	1	3	
Lower header	2	1	2	
Lower header cover	1	1	1	
Upper header	2	2	4	
Upper header cover	1	2	2	
Drain/lower header junction	1	1	1	

Table A.3-25 PANTHERS PCC Component Demonstration Test Matrix

Cycle Type	Number of Cycles	Maximum Pressure (kPa)	Maximum Temperature (Deg C)	Cycle Duration (Min.)
LOCA	10	379	Saturation	30
Pneumatic Test	300	758	Ambient	2

Table A.3-26 PANTHERS/PCC
LOCA Cycle Time History

PCC Inlet Pressure [kPag (psig)]	Time to Reach Pressure (Sec)
175 (25.4)	start*
249 (36.1)	<30
261 (37.8)	<65
379 (55)	<30 minutes

* The unit is initially pressurized with air at ambient conditions.

Table A.3-27 Isolation Condenser Structural Measurements

Measurement/Location	No. of Positions	Quantity at each Position	Total Measurements
Acceleration:			
Mid-length of tube	5	2	10
Drain line curve	1	3	3
Lower header cover	1	1	1
Upper header cover	1	5	3
Displacement:			
Steam distributor	1	1	1
Drain/lower header junction	1	1	1
Steam pipe lower zone	1	1	21
Total Strain:			
Inlet/upper header junction	1	6	6
Upper header/tube junction	5	1 or 2	7
Mid-length of tube	3	1	3
Tube/lower header junction	3	1	3
Lower header	2	2	4
Lower header cover	1	2	2
Upper header	2	4	8
Upper header cover	1	4	4
Drain/lower header junction	1	4	4
Drain line curve	1	2	2
Drain line/drain tube	1	4	4
Upper header cover bolts	3	2 or 1	5
Lower header cover bolts	3	2 or 1	5
Guard pipe/distributor	1	3	3
Support	1	2	2
Upper header near support	1	4	4
Permanent strain:			
Inlet/header junction	1	3	3
Upper header/tube junction	3	1	3
Lower header/drain junction	1	1	2
Temperature:			
Guard pipe/distributor	1	1	1
Inlet pipe/upper header	2	2	4
Upper header/tube junction	3	1	3
Tube/lower header junction	3	1	3
Lower header	2	1	2
Upper header	2	2	4
Drain line bend	1	1	1
Upper header cover	1	2	2
Lower header cover	1	1	1

Table A.3-28 IC Component Demonstration Test Matrix

Test Cond. No.	No. of Cycles	Cycle Type	Initial Pressure [mPag (psig)]	Inlet Pressure [mPag (psig)]	Initial Pool Temp. °C(°F)
1	1	1	9.480 (1375)	8.618 (1250)	<21 (70)
16	20	1	8.618 (1250)	8.618 (1250)	<32 (90)
17	5	4	8.618 (1250)	N/A	<32 (90)
18	1	5	9.480 (1375)	8.618 (1250)	<32 (90)

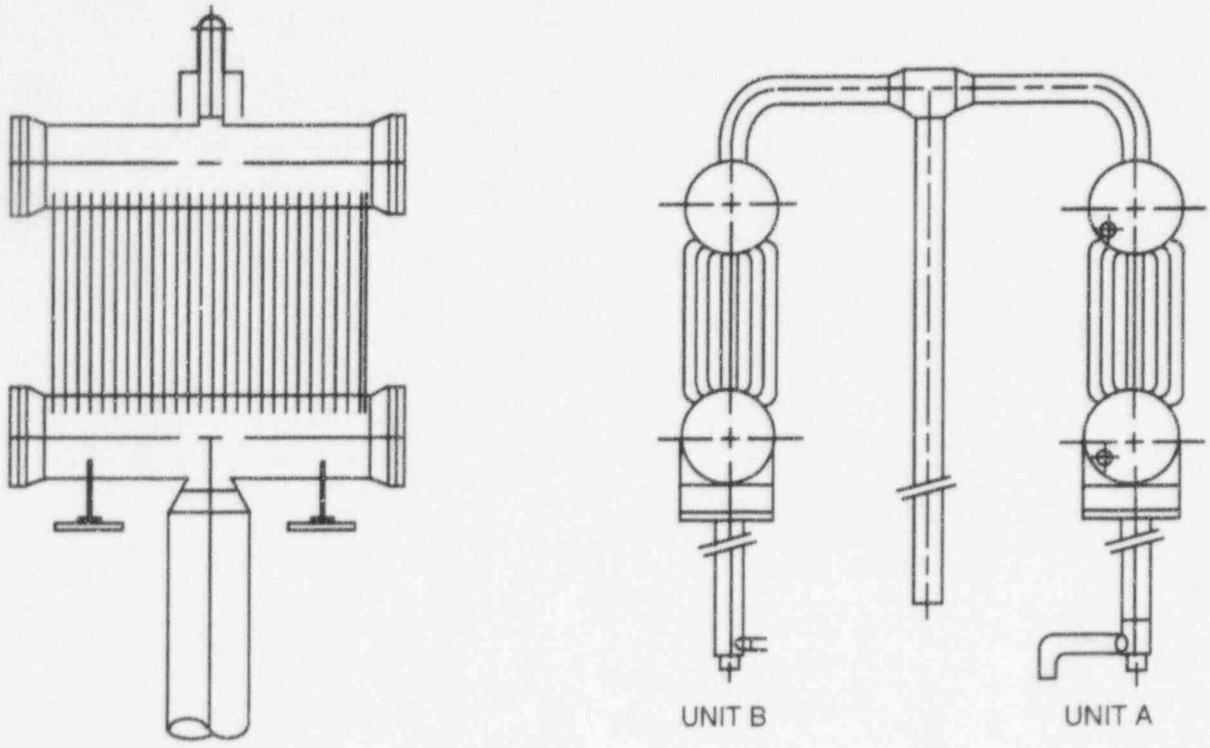

Figure A.3-1 Passive Containment Cooler Test Article

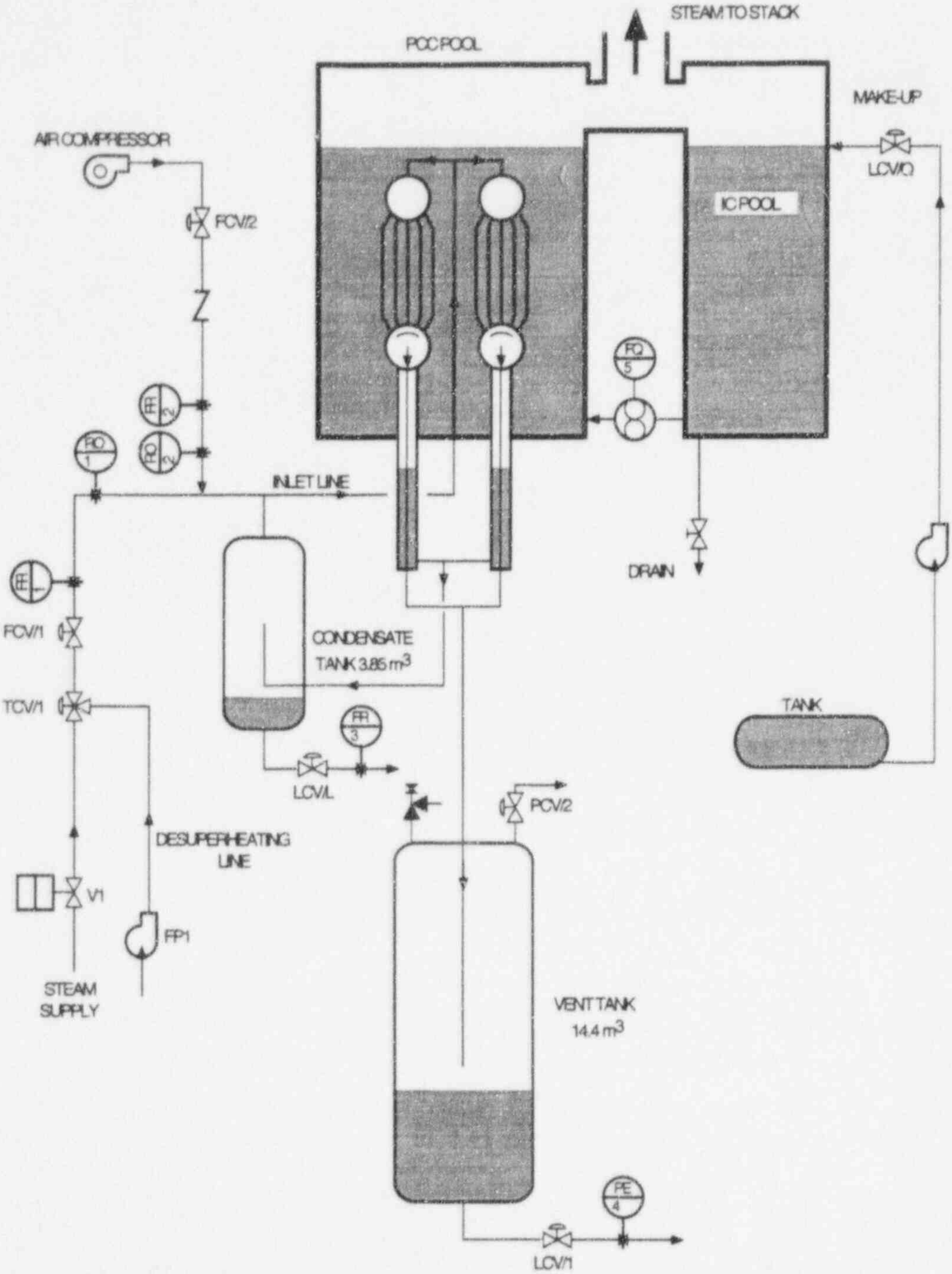

Figure A.3-2 PANTHERS/PCC Test Facility Schematic

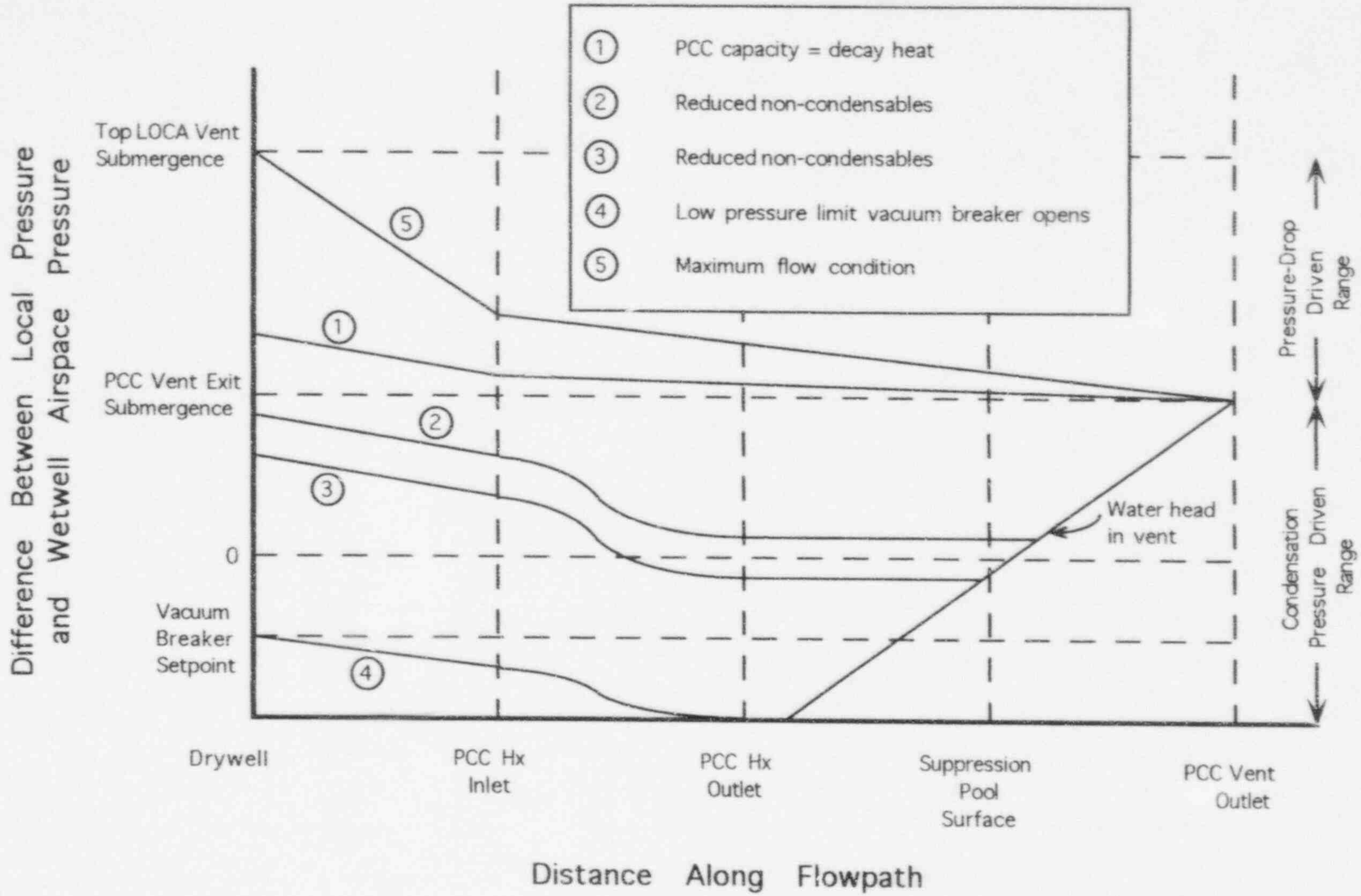

Figure A.3-3 PCC Heat Exchanger Operational Modes

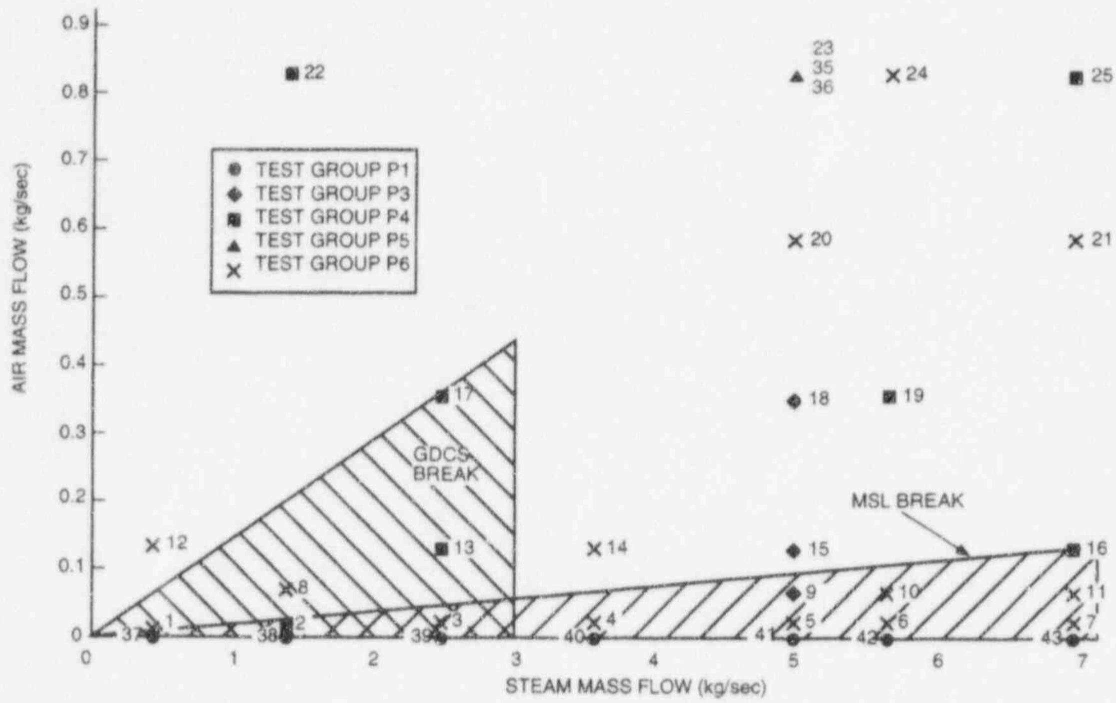

Figure A.3-4 Comparison of PANTHERS/PCC Steam-Air Test Range to SBWR Conditions

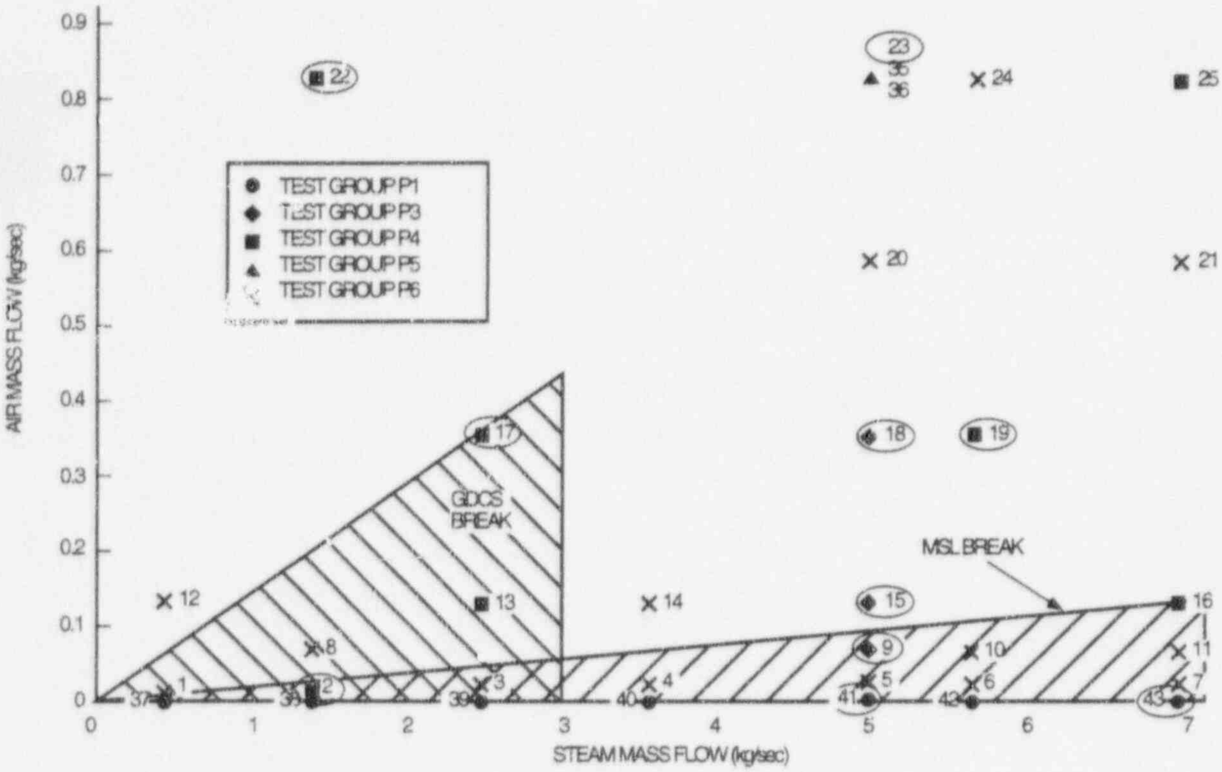

Figure A.3-5 TRACG PANTHERS/PCC Qualification Points

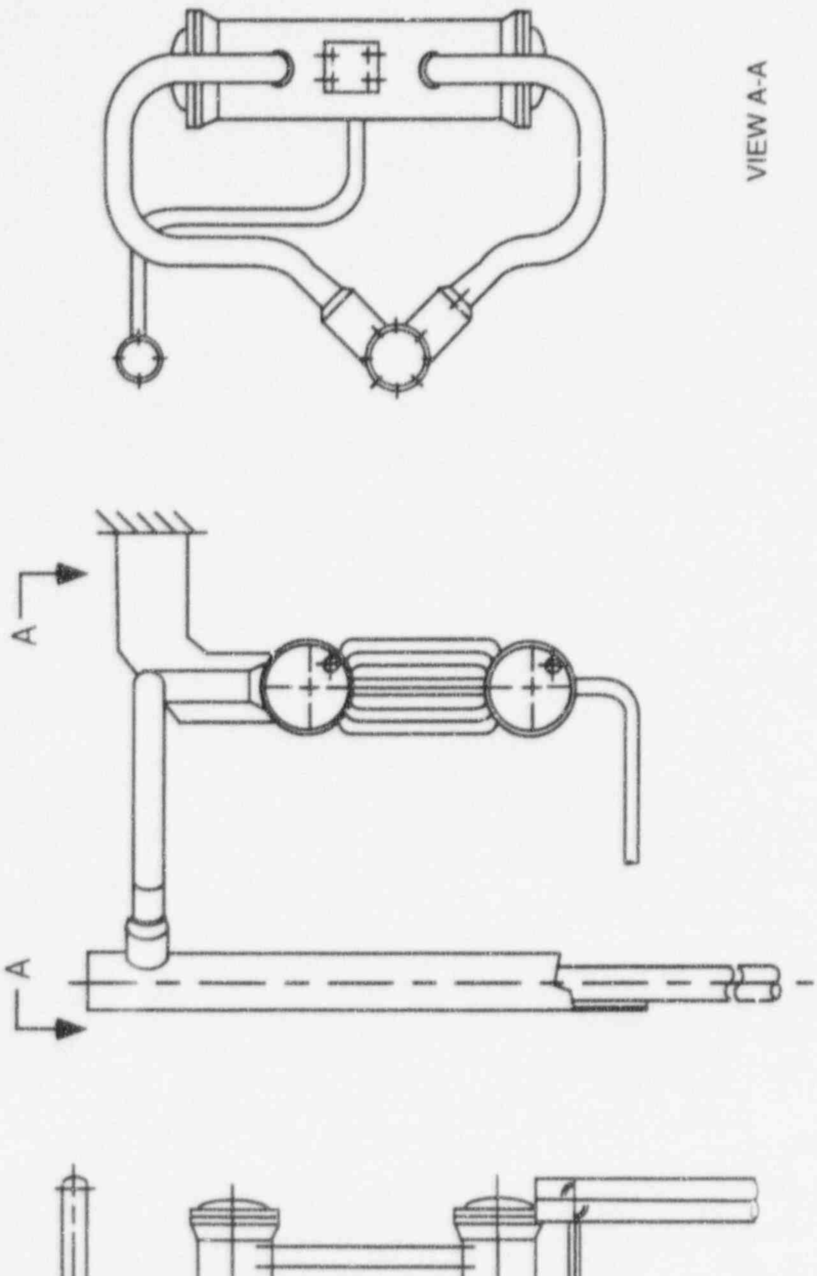

Figure A.3-6 Isolation Condenser Test Article

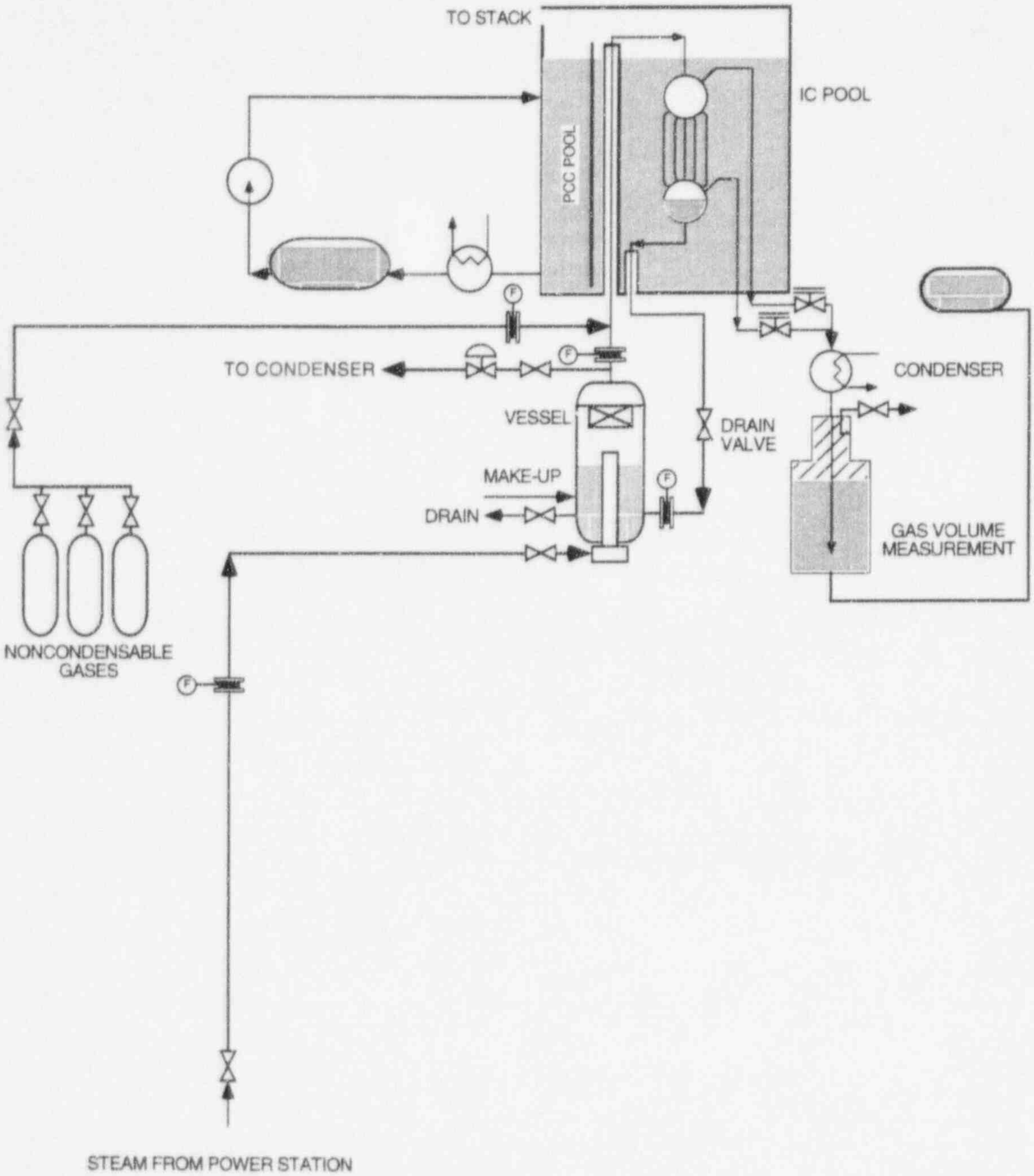

Figure A.3-7 PANTHERS/IC Test Facility Process Diagram

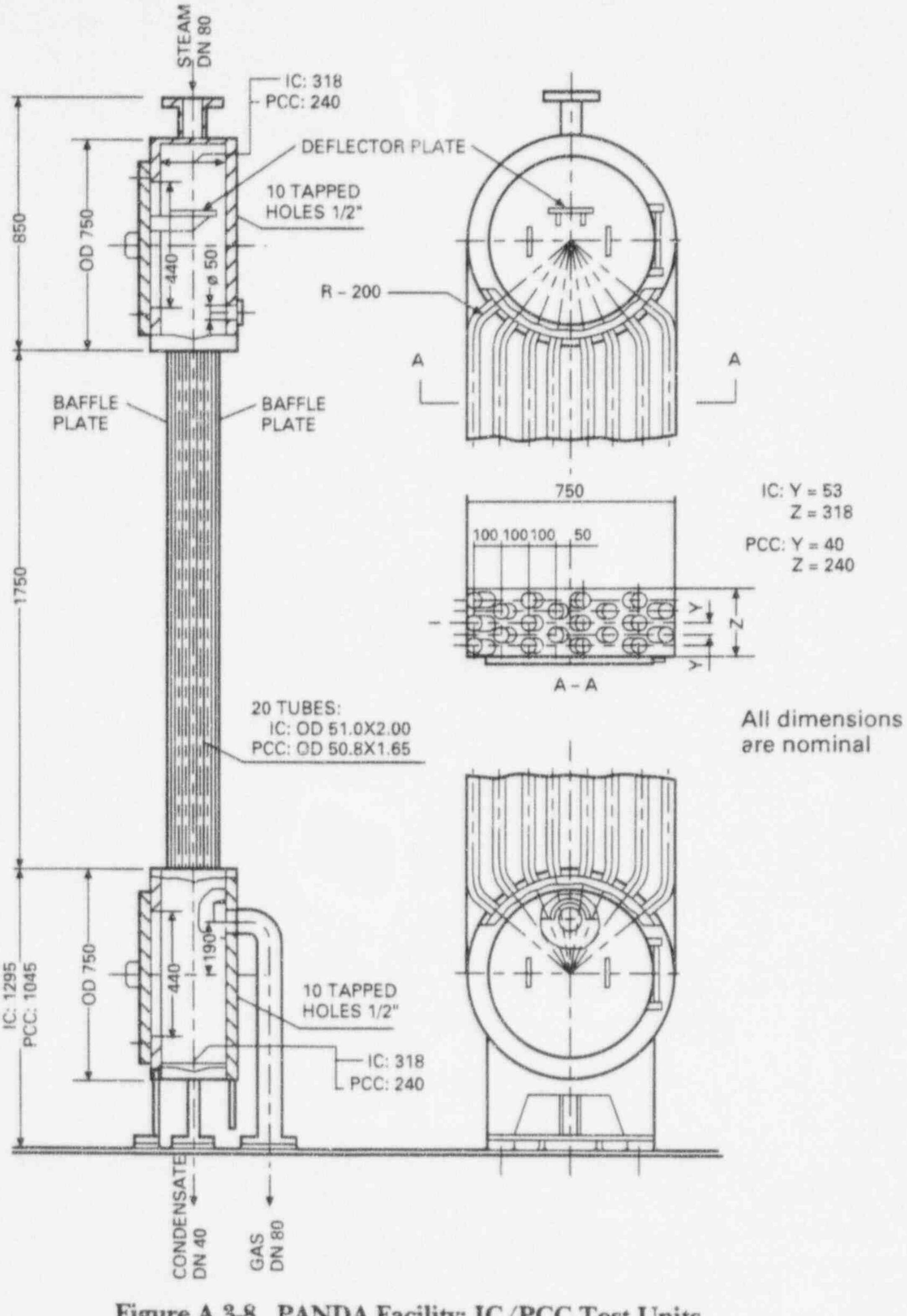

Figure A.3-8 PANDA Facility: IC/PCC Test Units

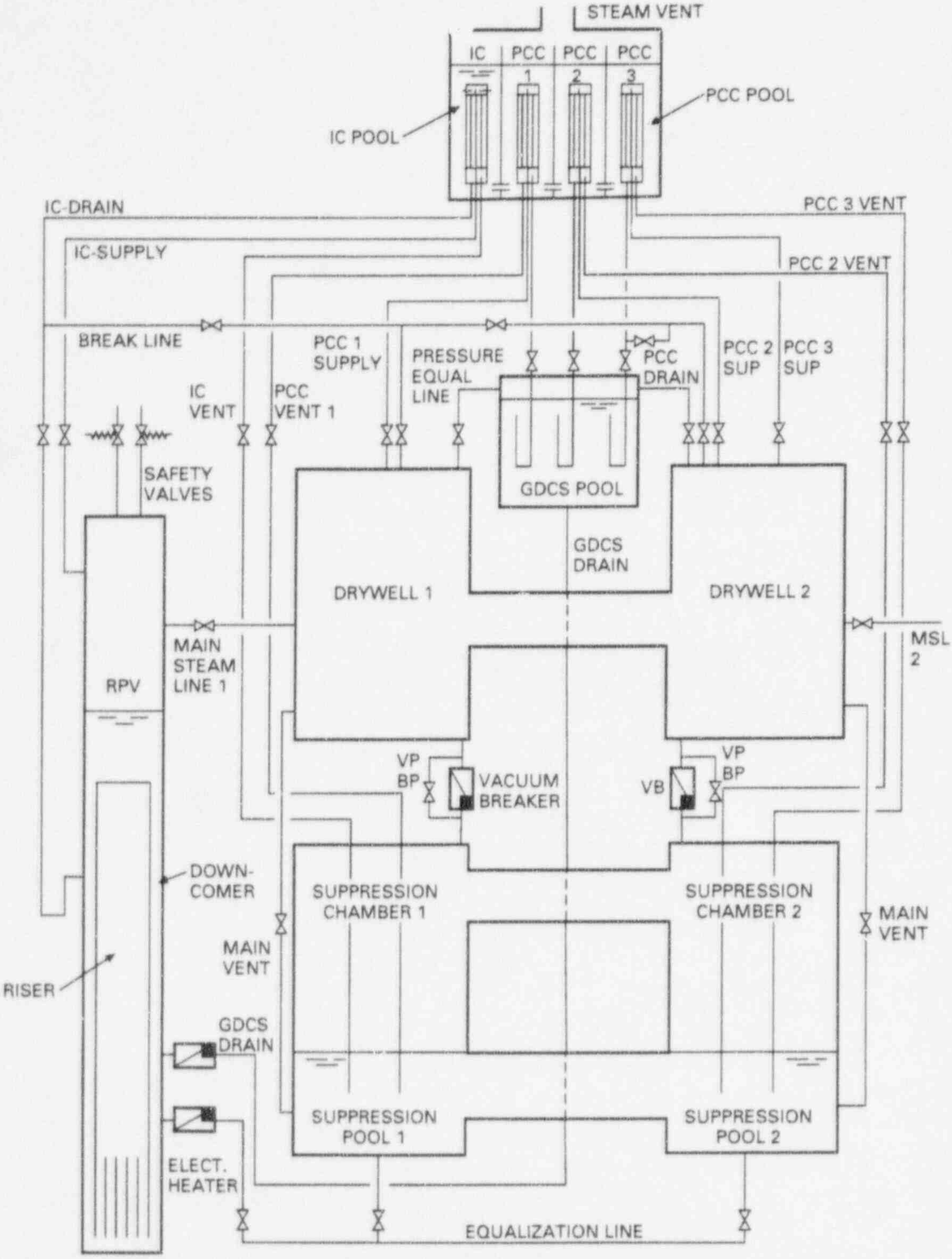

Figure A.3-9 PANDA Facility Schematic

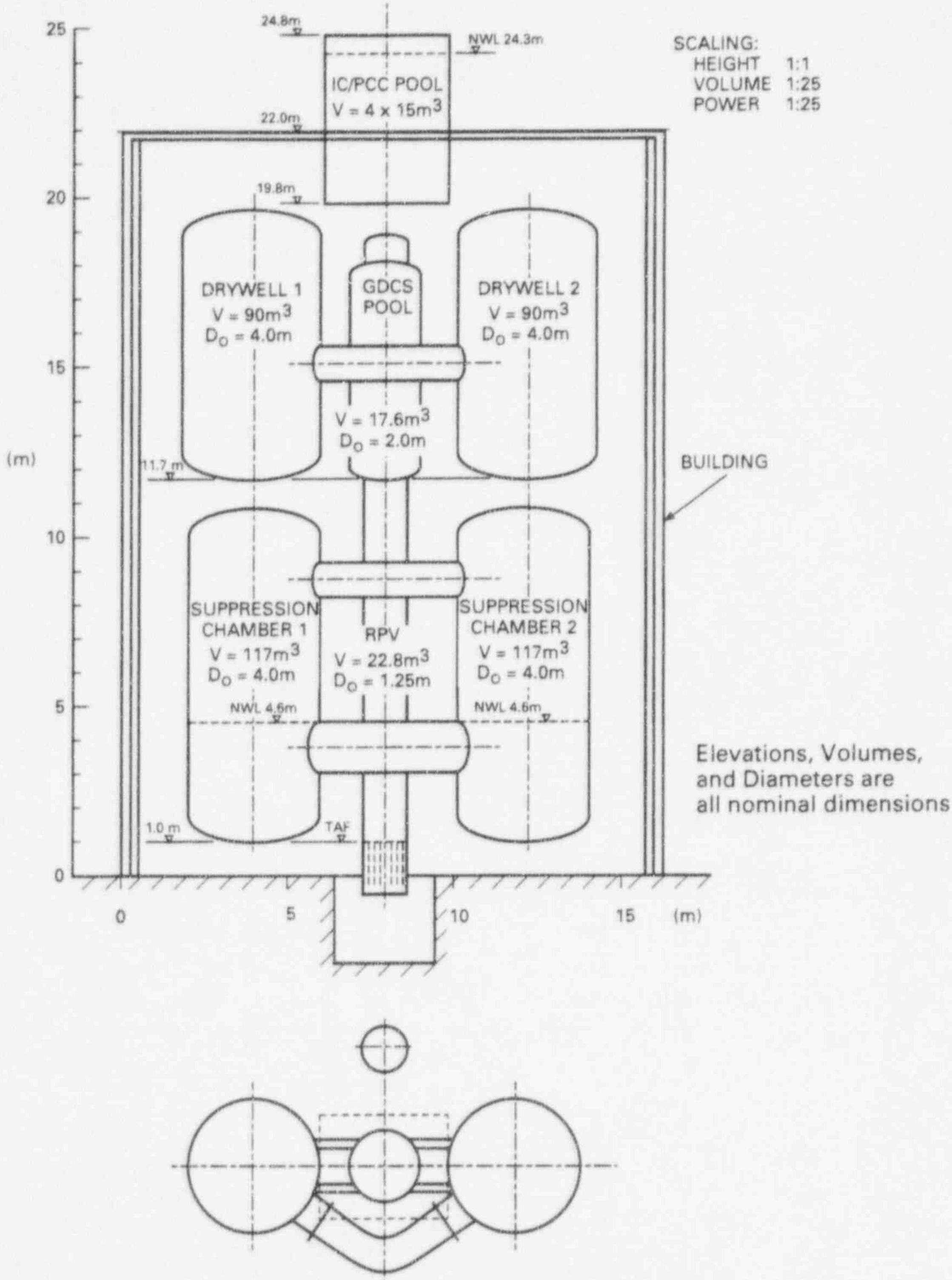

Figure A.3-10 PANDA Facility: Configuration of Vessels

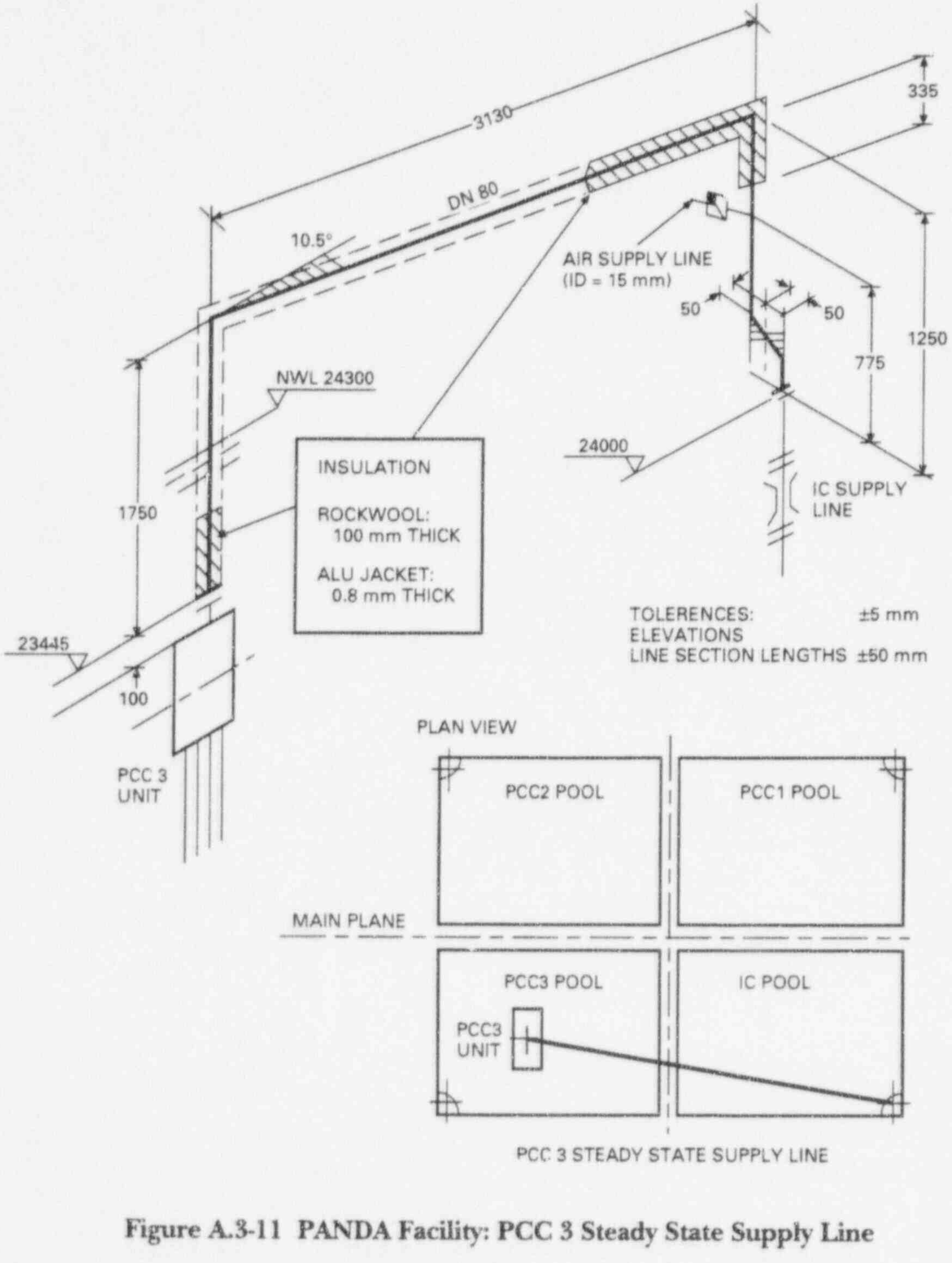

Figure A.3-11 PANDA Facility: PCC 3 Steady State Supply Line

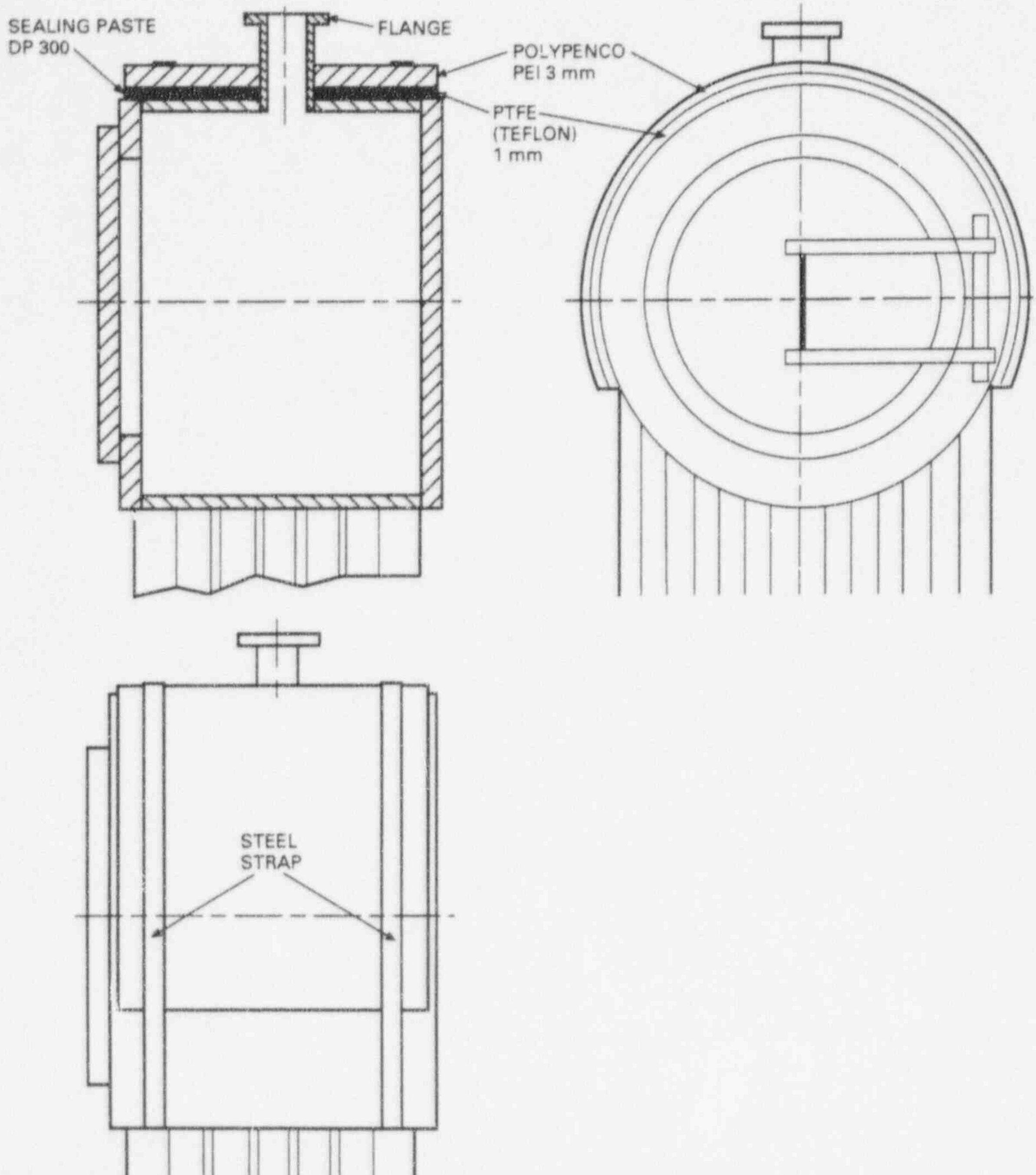

Figure A.3-12 PANDA Facility: IC/PCC Header Drum Insulation

Note: Lower Header Similar

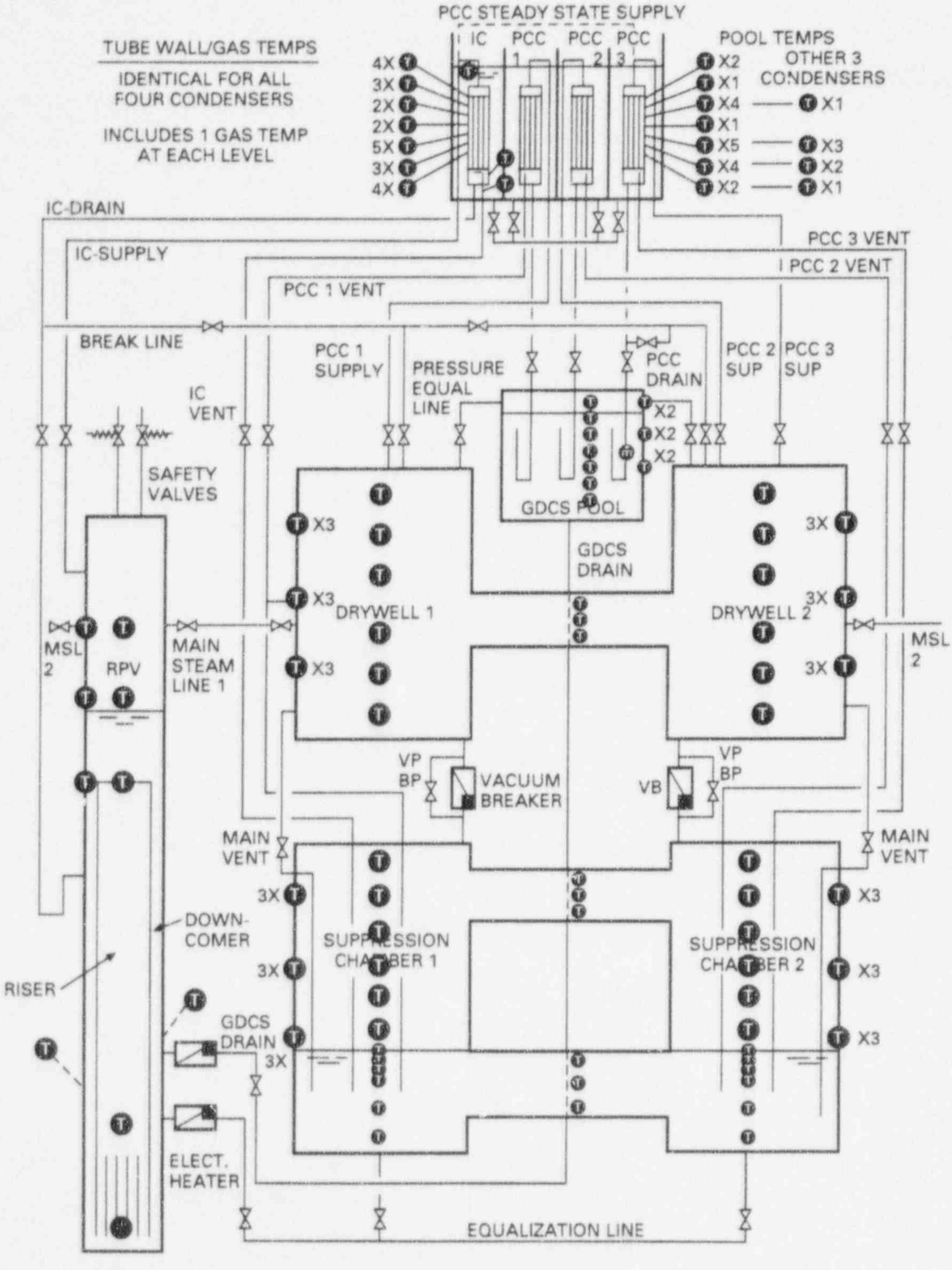

Figure A.3-13a PANDA Instrumentation: Condenser, Pool, and Vessel Temperatures

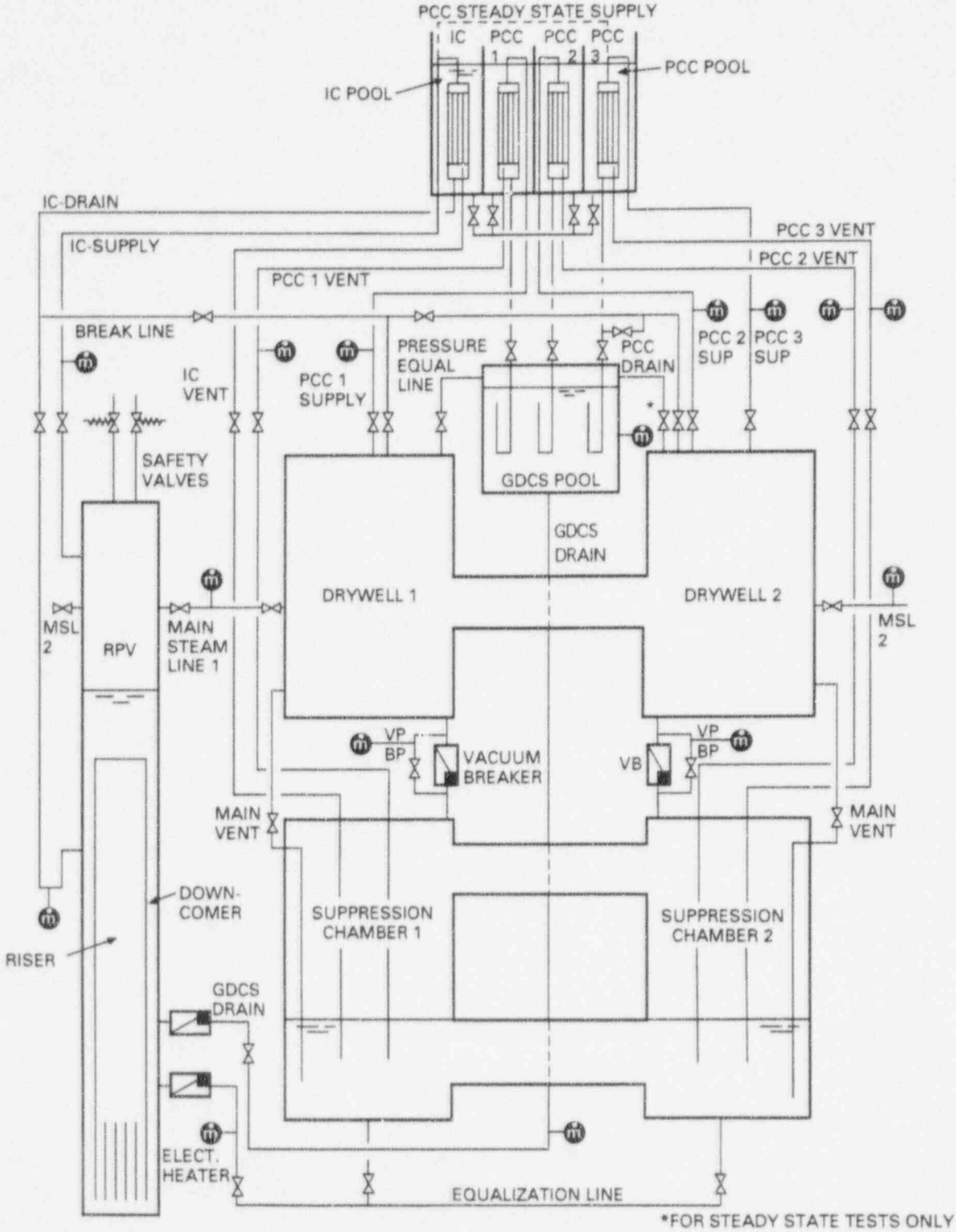

Figure A.3-13b. PANDA Instrumentation: Mass Flow Rates

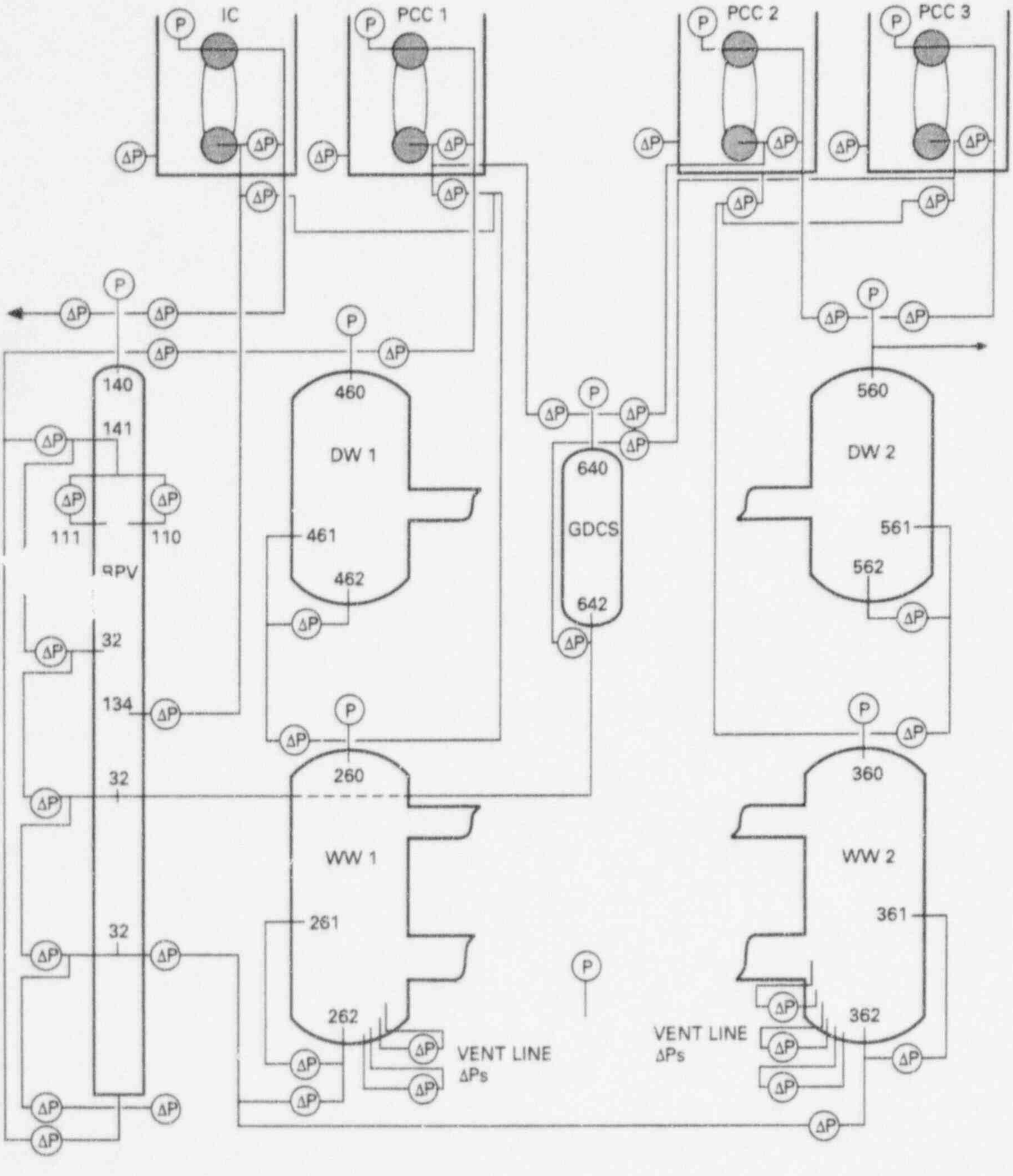

Figure A.3-13c PANDA Instrumentation: Absolute and Differential Pressures

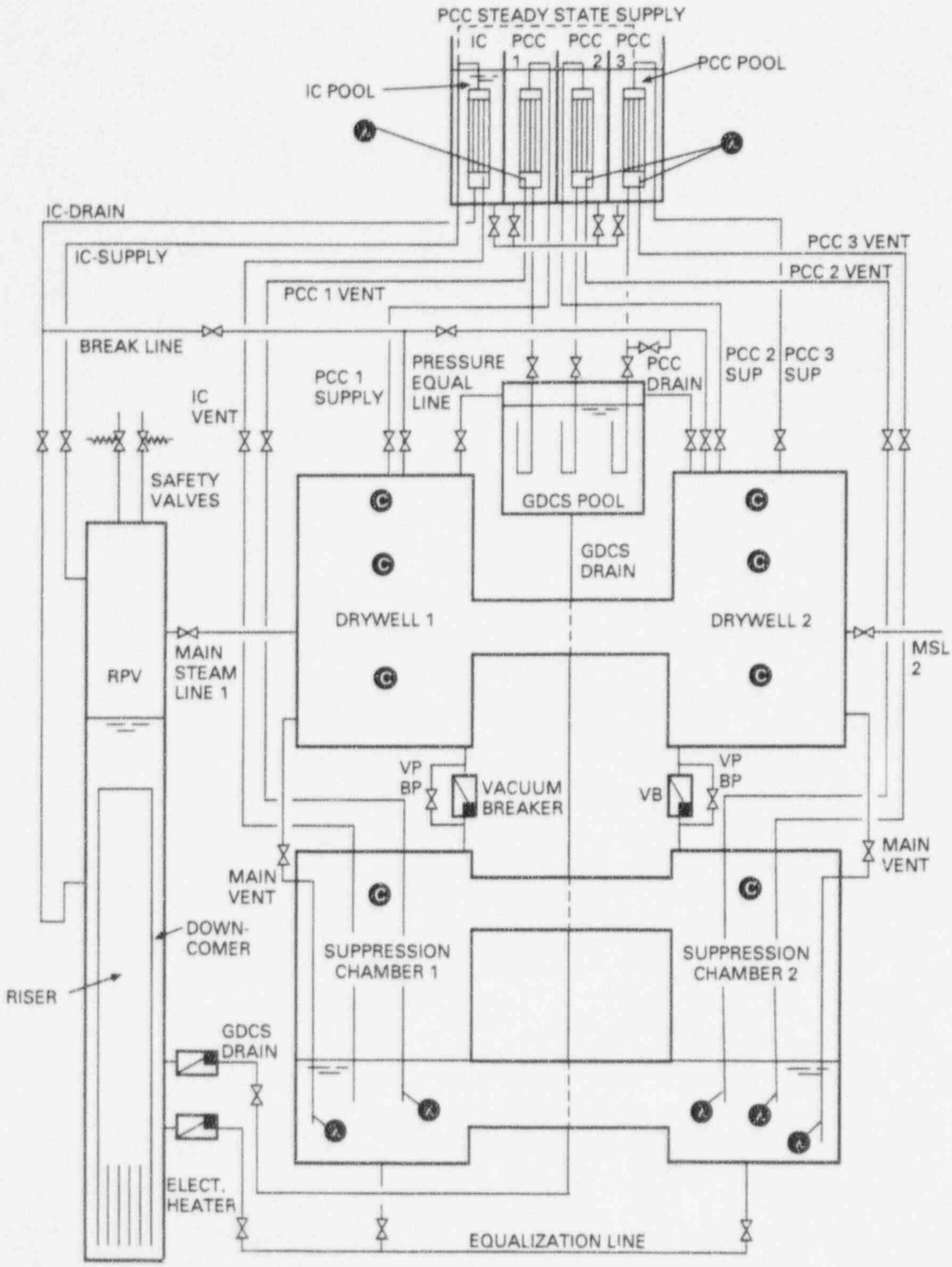

Figure A.3-13d PANDA Instrumentation: Non-Condensable(c) and Phase Detectors (λ)

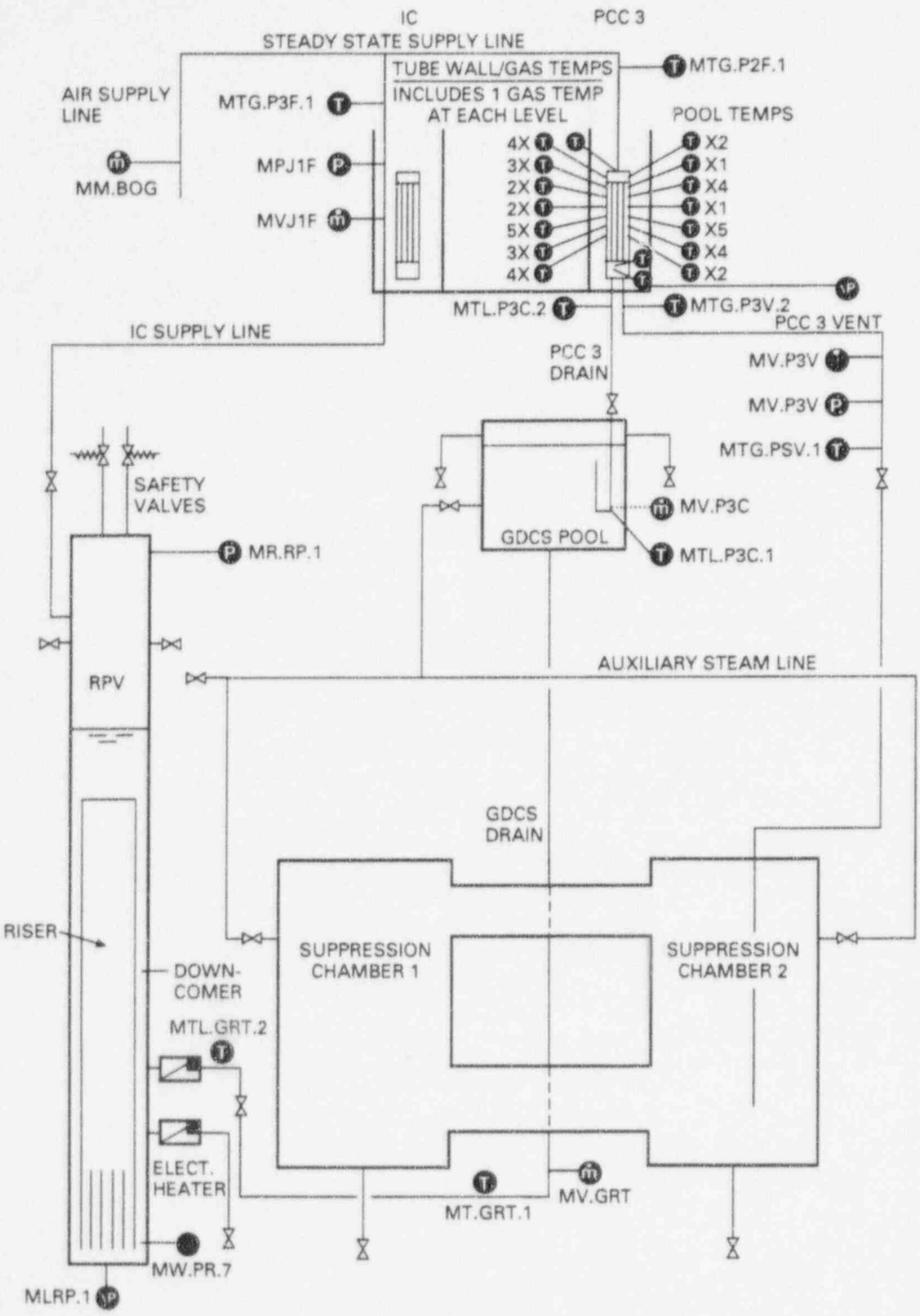

Figure A.3-14 PANDA Steady State Test Instrumentation/Configuration

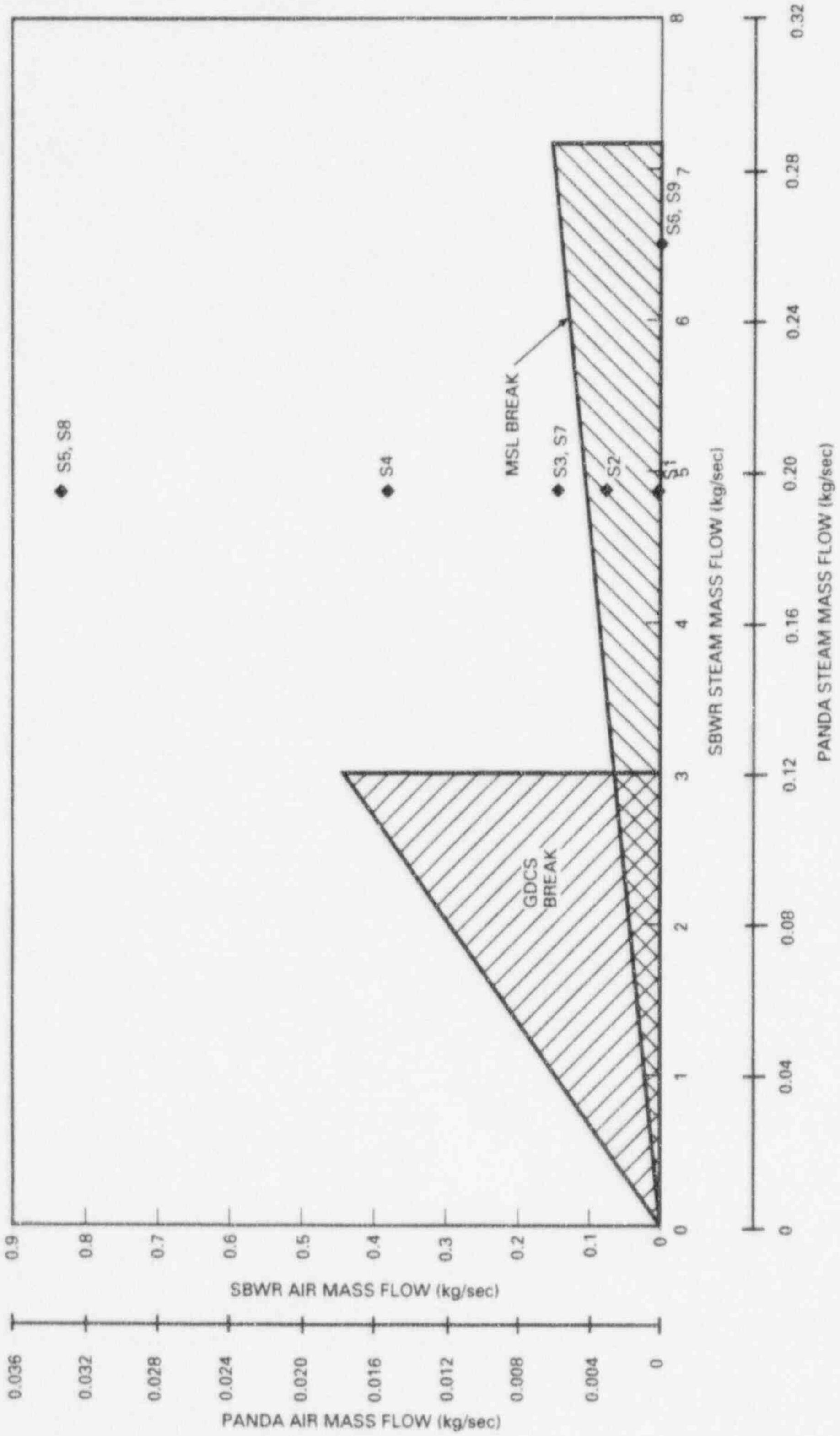

Figure A.3-15 Comparison of PANDA Steady-State Test Range to SBWR Condition

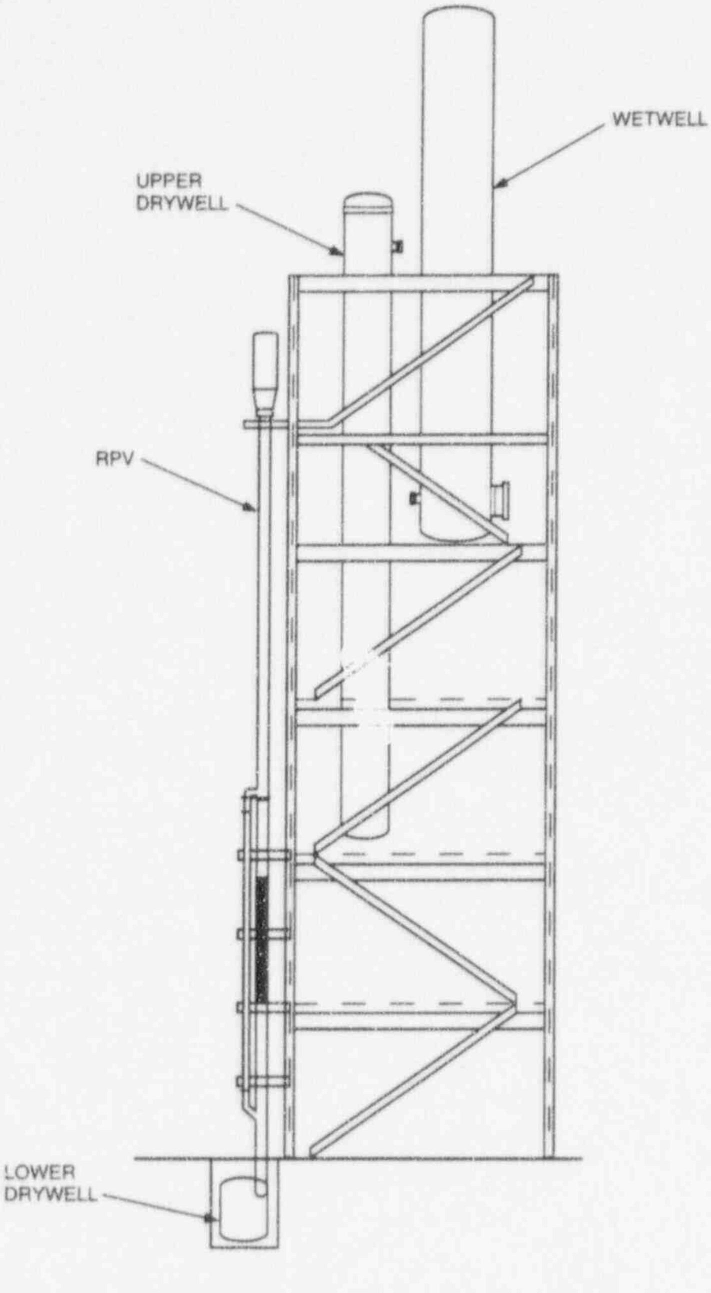

Figure A.3-16 GIST Facility Schematic

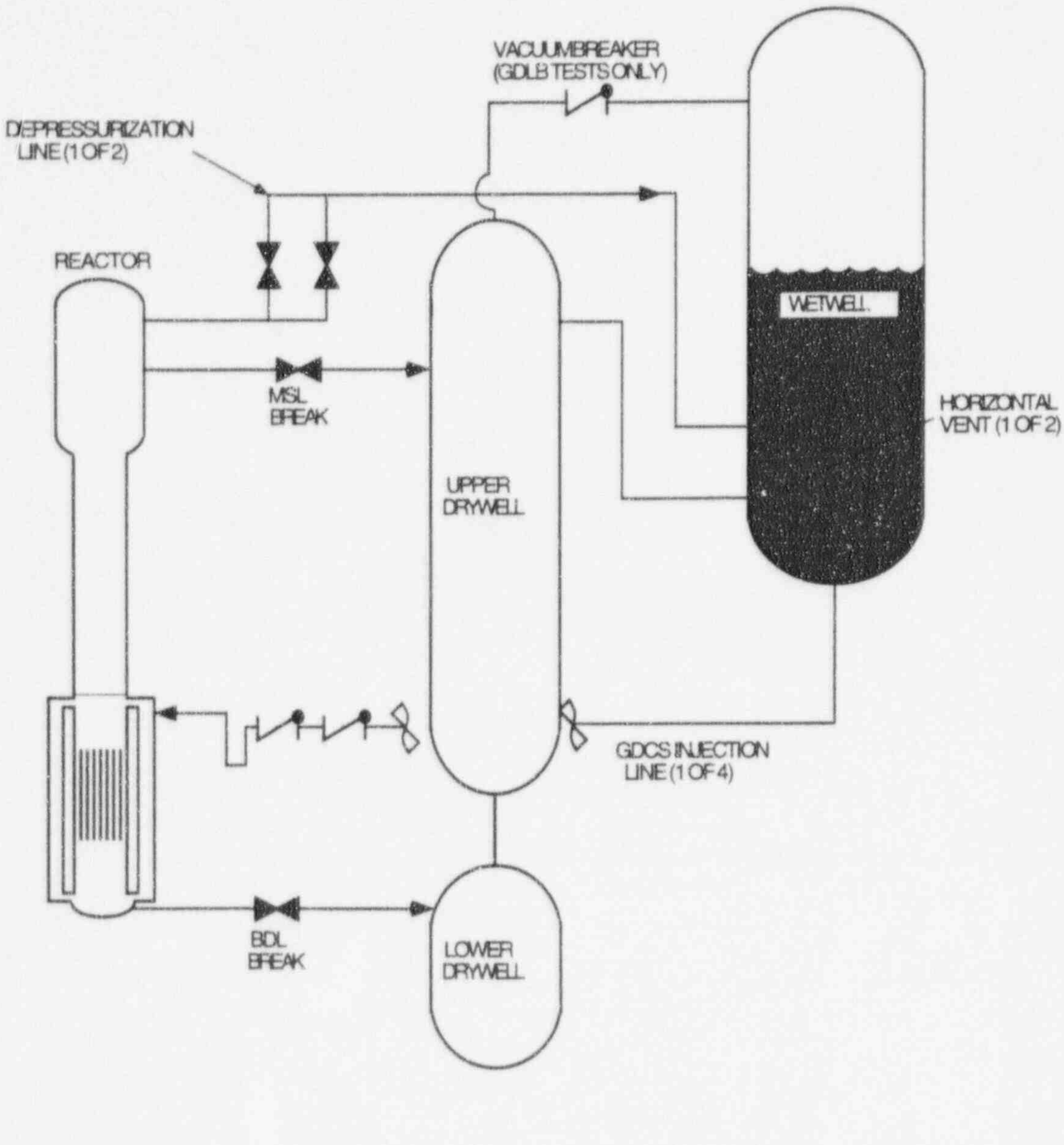

Figure A.3-17 GIST Facility Piping Arrangement

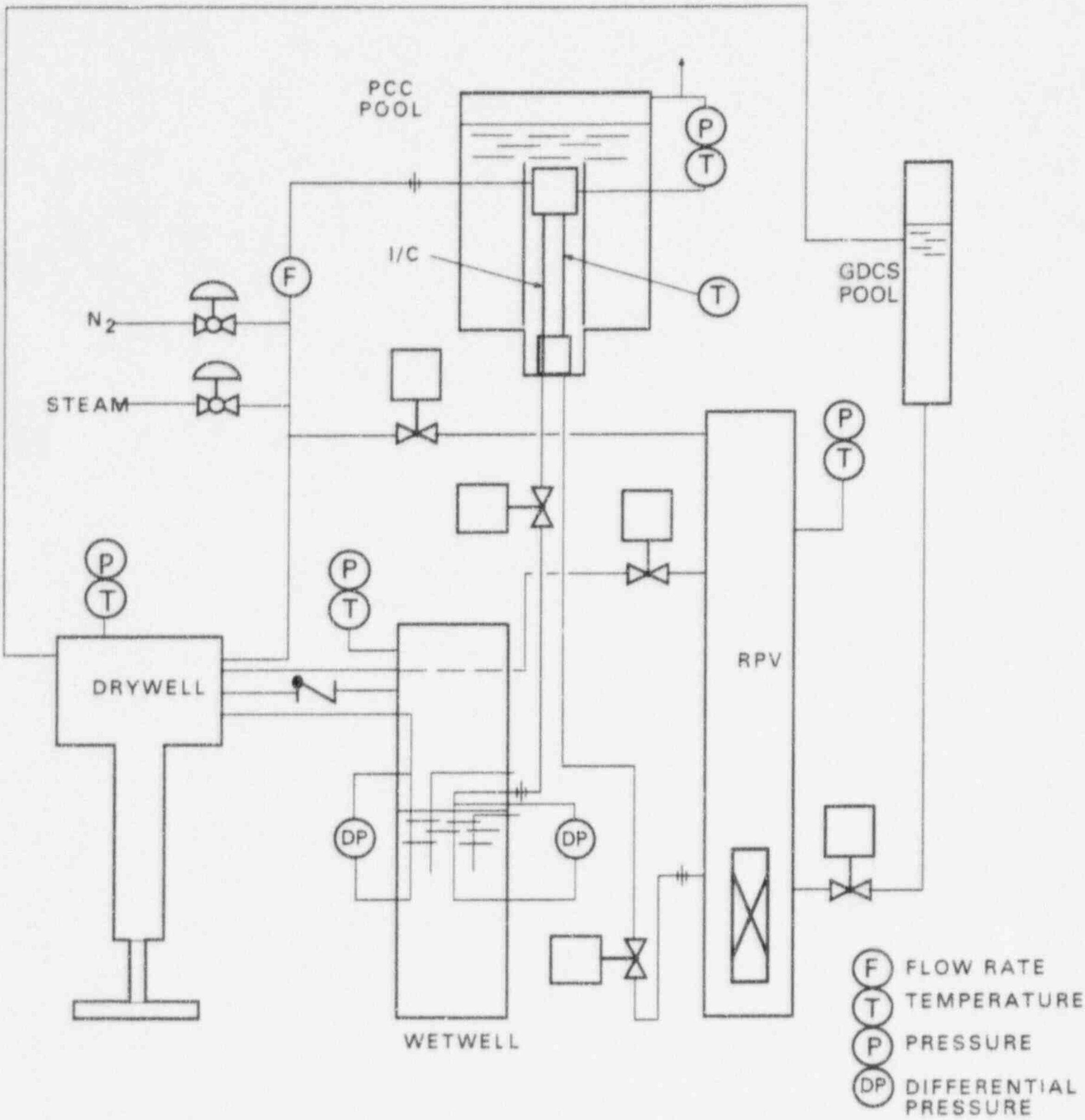

Figure A.3-18 GIRAFFE Test Facility Schematic (Phase 1)

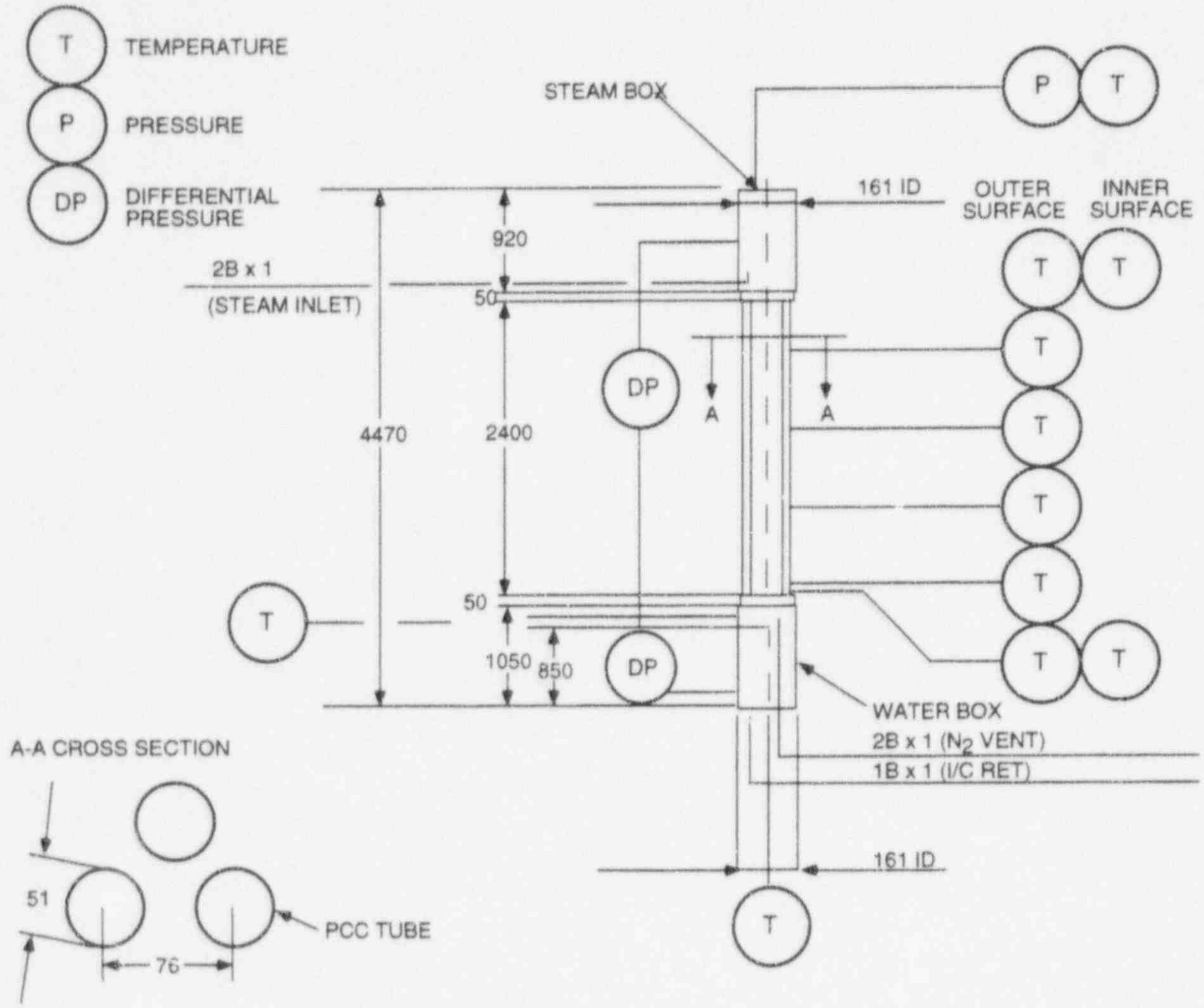

Figure A.3-19 GIRAFFE PCC Unit

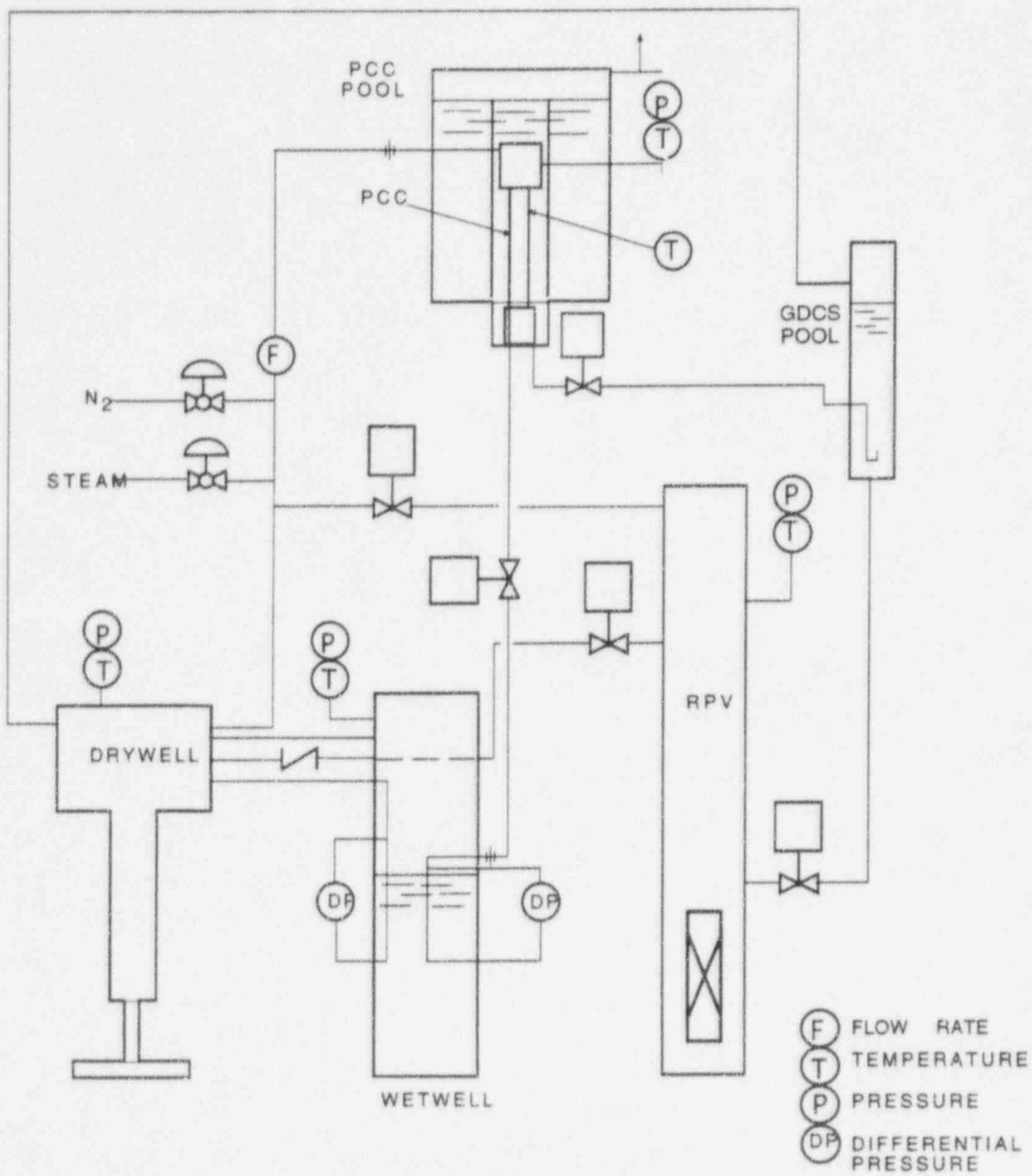

Figure A.3-20 GIRAFFE Test Facility Schematic (Post Phase 1)

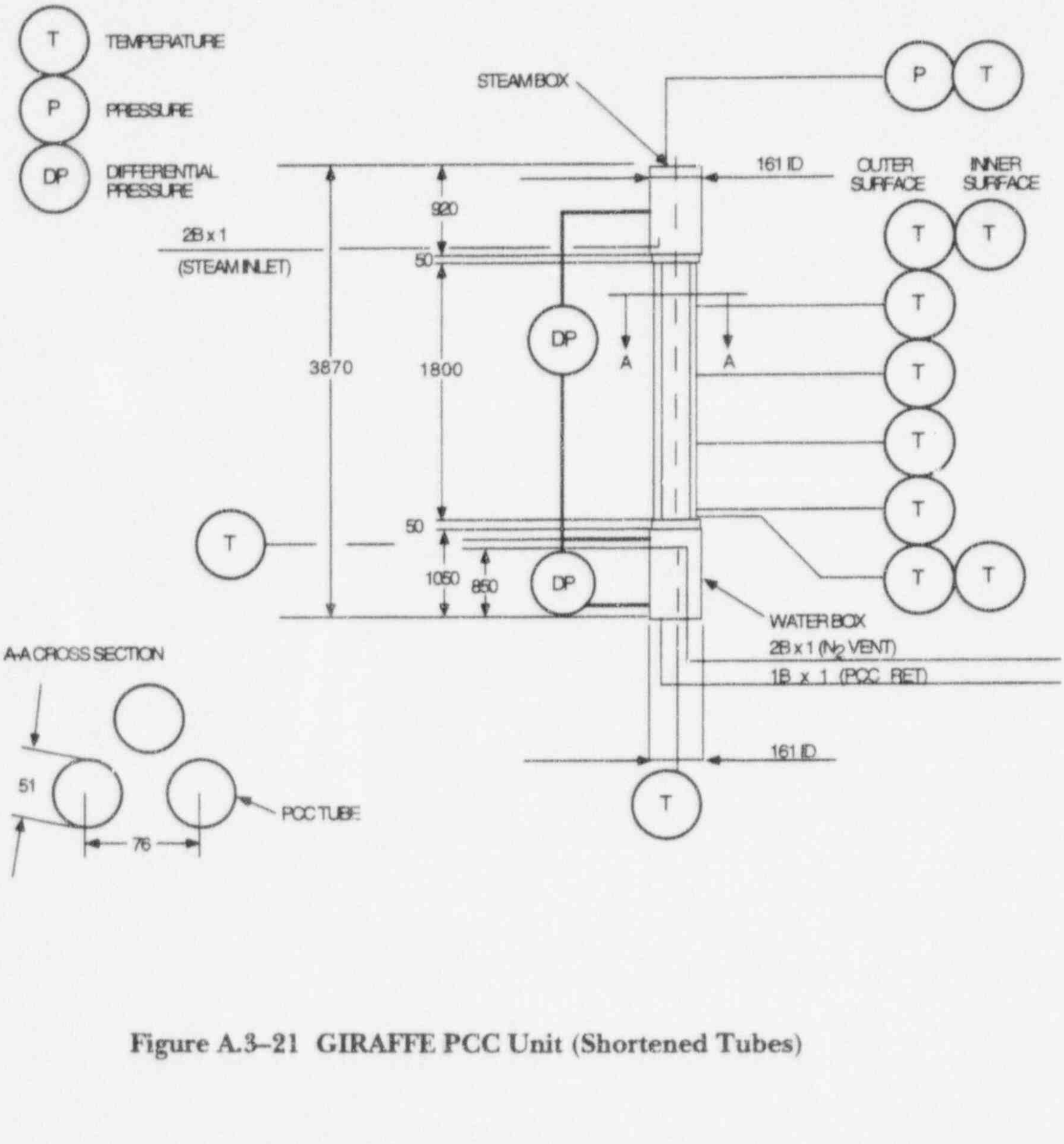

Figure A.3–21 GIRAFFE PCC Unit (Shortened Tubes)

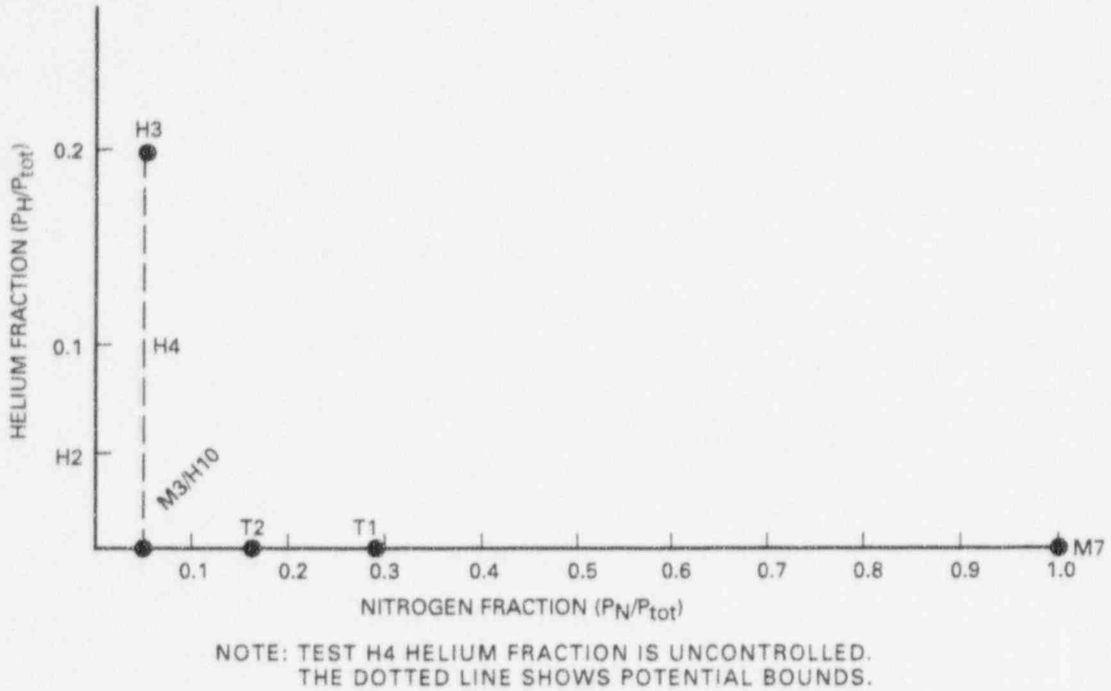

Figure A.3-22 PCC Startup Initial Condition Map

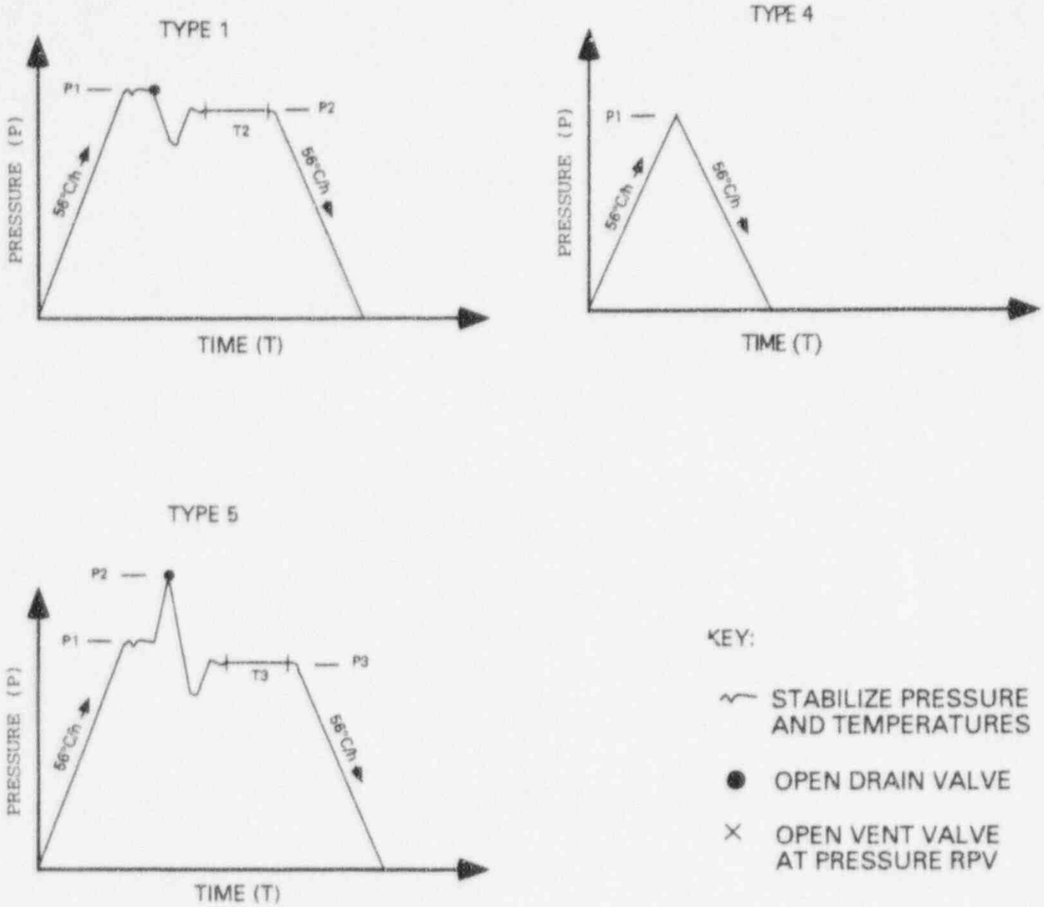

Figure A.3-23 IC Cycle Types

ATTACHMENT A1 - TABLE OF CONTENTS FOR TEST AND ANALYSIS DOCUMENTS

Apparent Test Results

- Brief report on each test
- Tables and plots of key measurements
- Identification of any non-conformances related to test results

Data Transmittal Report

1.0 Introduction
- General description and purpose of tests
- Purpose of report

2.0 Objectives
- General Objectives
- Specific Objectives

(Note: General Objectives are given in TAPD, Appendix A, for each of the tests)

3.0 Test Facility Description
- Detailed description of facility layout
- Scaling study

(Note: Facility descriptions will be from the Test Specification and/or Test Plan and Procedures. Scaling study will be a reference to the final version of TAPD Appendix B.)

4.0 Instrumentation
- Instrument type and characteristics
- Calibration

5.0 Data Acquisition System
- Hardware configuration
- Data Reduction
- Software

6.0 Test Matrix
- Grouped by type of test

7.0 Test Results
- Grouped by type of test

8.0 Conclusions
- Adequacy of test data
- Applicability to test objectives

9.0 References

Appendices

A. Instrument List (Type of instrument, number of instrument, measurement, and range)

B. Modified and Failed Instruments
- Listed by test

C. Facility Characterization Tests
- Pressure drop tests
- Heat loss tests

D. Error Analysis
- Maximum error of measurement

E. Data Records
- Format of Data Tapes

Data Analysis Report

1.0 Introduction
- General description and purpose of tests
- Purpose of report

2.0 Objectives
- General Objectives
- Specific Objectives

(Note: General Objectives are given in TAPD, Appendix A, for each of the tests)

3.0 Test Analysis
- Grouped by type of test
- Description of test conditions
- Analysis of test results

(Note: Framework of test results analysis is given in the "Test Matrix and Data Analysis sections of the TAPD," Appendix A)
- Discussion of observed phenomena

4.0 Conclusions
- Adequacy of test data
- Applicability to test objectives

5.0 References

Preliminary Validation Results

1.0 Introduction

- General description and purpose of tests
- What tests will be used for assessment
- How data will be used

2.0 Brief description of Test Facility and Test Matrix
- Referenced to appropriate test reports

3.0 Applicability of data to SBWR
- Range of relevant parameters/ scaling groups compared to SBWR

4.0 TRACG model and nodalization
- Noding used and basis
- Any modifications for post-test analysis
- Justification for difference in nodalization vs. SBWR nodalization, if any
- Discussion of new models, if any

5.0 Test Simulation
- Choice of tests to be simulated
- Procedure for simulation, including initial and boundary conditions

6.0 Qualification results - data vs. predictions
- Comparisons between data and TRACG results
- Plots and discussion of key parameters for each test

7.0 Results of Assessment
- Adequacy of TRACG models
- Implications for SBWR calculations

8.0 References

TRACG Computer Code Qualification For SBWR

Abstract

1.0 Introduction
1.1 Relationship to CSAU Process

2.0 Qualification Strategy
2.1 Assessment Matrix
2.2 Coverage of PIRT Phenomena

3.0 Separate Effects Tests
- Refer to previous report NEDE-32177P

4.0 Component Performance Tests
4.1 PANTHERS PCC Performance
4.2 PANTHERS IC Performance

- 4.3 PANDA PCC Performance
- 4.4 Suppression Pool Stratification in Blowdown Tests
- 4.5 SLCS Accumulator Performance

5.0 Integral System Tests
- 5.1 GIST tests (from NEDE-32177P)
- 5.2 GIRAFFE/Helium Tests
- 5.3 GIRAFFE/SIT
- 5.4 1/6 Scale Boron Mixing
- 5.5 CRIEPI Geysering Tests
- 5.6 PSTF MARK III Containment Response
- 5.7 4T / MARK II Containment Response
- 5.8 Dodewaard Startup
- 5.9 PANDA Transient Tests M2 - M10

6.0 SBWR Plant Nodalization
- 6.1 Reactor Vessel
- 6.2 Containment

7.0 Determination of Model Uncertainties and Bias
- 7.1 Application to LOCA/ECCS and transients
- 7.2 Application to containment

8.0 Conclusion
- 8.1 Adequacy of TRACG models
- 8.2 Implications for SBWR calculations

APPENDIX B - SCALING APPLICABILITY

B.1 Introduction

This appendix contains a discussion of the scaling analyses which show that the SBWR thermal-hydraulic test facilities – PANTHERS, PANDA, GIST, and GIRAFFE – are scaled appropriately to meet the objectives outlined in Appendix A. Additionally, a scaling analysis of the CRIEPI Natural Circulation Thermo-Hydraulic Test Facility for start-up and rated conditions has been added.

This appendix is an extension of the information contained in "Scaling of the SBWR Related Tests" (Reference 32). It contains an expansion of the Bottom-Up scaling parameters developed in that report and the quantitative details of applying the scaling method to the four major SBWR test facilities – PANTHERS, PANDA, GIST, and GIRAFFE. Material developed or discussed in Reference 32, in general, is not repeated here.

The scaling reported in Reference 32 follows the Hierarchical, Two-Tiered Scaling (H2TS) methodology outlined in Reference 39. Before presenting numerical comparisons of the SBWR and scaled test facilities, it is important to understand what level of differences between the two is acceptable. As noted in Reference 32:

> "System tests (such as the GIST, GIRAFFE and PANDA tests) do not have to provide exact system simulations of the prototype. In fact, it is neither practical nor desirable to attempt to provide such exact simulations. However, system tests do provide data covering all essential phenomena and system behavior under a variety of conditions, which are used to qualify a system code (in this particular case, the TRACG code used for safety analysis by GE).
>
> To obtain data in the proper range of systems conditions, the relative importance of the phenomena and processes present in the tests should not differ significantly from what is expected to take place in the SBWR. Similarly, the overall behavior of the test facility should not diverge significantly from that of the SBWR; in particular, one should not observe bifurcations in the system behavior leading to quite different intermediate or end states. Finally, the test should provide sufficiently detailed information, obtained under well-controlled conditions, to provide an adequate and sufficient database for qualifying a systems code, TRACG."

This is a good qualitative discussion of how "prototypic" the scaling of the test facilities must be, but it does not result in a specific quantitative rule for determining acceptable deviations. In fact, no such rule can exist, since some parameters are more important than others and bifurcations do not occur at a specific level of distortion. Because of this, each parameter must be handled on an individual basis using engineering judgment to determine the acceptability of the scaling. The approach used is to group parameters into those that are particularly important to the system behavior and those that are not, with particular emphasis placed on the first group. In addition, a discussion of less important parameters having particularly poor scaling is included.

Both Top-Down and Bottom-Up scaling are performed as part of the analysis. Top-Down scaling is an inductive system approach that results in scaling parameters for each of the volumes and flow paths in a system. Local phenomena within a volume are not considered within the Top-Down scaling. Bottom-Up scaling is a deductive process-and-phenomena approach that considers the specific phenomena that may occur within a region.

B.2 Integral System Tests

B.2.1 Application to Test Facilities

In applying the scaling equations to the SBWR and test facilities, a single point in time during a single event was selected, the beginning of the test simulation for a Main Steam Line Break (MSLB).

The reference values used in the scaling parameters are taken from the supporting documentation for each of the facilities and the SBWR. When values are not explicitly given, calculations are used to determine the needed value.

The details of how the generic scaling equations are used to calculate the final values for the facilities are contained in Attachment B1. Attachment B1 indicates the reference values used to calculate the scaling parameters for those that are not obvious. Selection of proper reference values is important in order to obtain meaningful results with the scaling parameters. The attachment contains tables with the complete set of scaling parameters for each of the tests. Parameters that are particularly important to each of the tests are indicated by grey shading of the numbers. A scaling parameter with a large distortion, whether important or not, is indicated by a circle around its value. Both of these sets of numbers are summarized in tables in the following sections.

B.2.2 Scaling of GIST Facility

B.2.2.1 Facility Description and Test Characteristics

The GIST facility is a full-vertical-scale, multi-component integrated system test as described in Reference 42.

The facility, having a system scale of 1:508, is composed of the following regions:

- Reactor Vessel
- Upper Drywell
- Lower Drywell
- Wetwell/GDCS pool

Note, the IC and PCC systems are not represented.

There are two substantial differences in configuration between the GIST facility, which represented an early SBWR design, and the final SBWR design as referenced in the SSAR. First, the GIST GDCS pool is combined with the suppression pool and located in the wetwell, rather than being a separate pool located in the drywell, as in the SBWR SSAR design. Second, all of the RPV depressurization in GIST occurs via SRVs that exhaust to the suppression pool rather than the combination of SRVs (exhausting to the suppression pool) and DPVs (exhausting to the drywell) currently used in SBWR. In addition, there are several scaling differences between the GIST facility and SBWR as discussed below. The largest of these results because the GDCS design flow rate has been increased substantially since the GIST tests. A complete discussion of the differences is contained in the appendix of Reference 42.

B.2.3 Scaling of GIRAFFE Facility Phase 1 and Phase 2 Tests

B.2.3.1 Facility Description and Test Characteristics

The GIRAFFE facility is a full-vertical-scale, multi-component integrated containment system test with a system scale of 1:400.

The facility is composed of the following regions:

- Reactor Vessel
- Drywell
- Wetwell
- GDCS pool
- PCC/IC Unit
- PCC/IC pool

The GIRAFFE facility was designed and constructed such that the single, three-tube IC/PCC could be operated as either an isolation condenser or a passive containment cooler. For the IC/PCC series of tests described in Reference 43 and in Appendix A, Subsection A.3.1.5 the unit was configured as a PCC. Additionally, the GIRAFFE facility can be configured to operate as either a LOCA integral systems test, or as a steady state performance test of the IC/PCC unit.

The Phase 1, Step 1 IC/PCC tests were steady-state performance tests which yielded support information on the heat rejection rate of a PCC. This data is used to corroborate PANDA and PANTHERS PCC component performance at a third scale. During the Phase 1, Step 3 Main Steam Line Break Test and the Phase 2 Main Steam Line Break Test, the facility was in the integral system test configuration. It is this configuration that is of primary interest; scaling for this configuration is discussed here.

Two changes in the facility configuration were made between the Phase 1, Step 3 and the Phase 2 tests: (1) adding micro-heaters to the outside walls of the various vessels to reduce heat losses to the environment, and (2) rerouting of the PCC condensate return line to the GDCS pool instead of the RPV, consistent with the SBWR SSAR design.

The GIRAFFE integral systems tests are used to assess system performance during the Long-Term PCCS period of a LOCA. This covers the time frame from approximately 1 hour after LOCA onward as shown in Figure 5.3-1. By this time in the event, the reactor has been depressurized and the GDCS has nearly or completely drained. The GDCS pool is acting as a collector for condensate returning from the PCCS, and draining it to the RPV. The SRVs and DPVs have little influence during this stage of the event since depressurization is complete.

B.2.4 Scaling of GIRAFFE Facility for Systems Interaction Tests (SIT)

B.2.4.1 Facility Description and Test Characteristics

The GIRAFFE/SIT integral systems tests are used to assess system performance during the late blowdown, GDCS and long-term PCCS periods of a LOCA.

Several modifications to the GIRAFFE facility were made since the Phase 2 tests. A second IC/PCC is available so both the IC and PCC can be simulated at the same time. Also, the line orifices will be modified to better represent the current SBWR design. The heat loss calibration will be repeated and the method of controlling the micro-heaters will be modified. The specific details of these modifications will be included in Appendix A of Revision C of this report. The test configuration will be similar to that of the Phase 2 tests shown in Figure A.3-18 of Appendix A.

Since these tests will be performed in an existing test facility, many of the parameters affecting scaling are established already. The purpose of this section is to characterize the distortions that may occur as a result of the fixed parameters, and to determine how to best scale the remaining parameters to provide the best fidelity with the SBWR design. The facility heights, volumes, general piping configuration, and heat loss characteristics are set by the existing facility. The Initial and Boundary conditions, together with the piping orifices, will be adjusted to provide the best fidelity possible with this facility.

B.2.4.2 Top-Down Scaling

The Top-Down scaling is used to assure that the overall global characteristics of the test behavior match those of the SBWR. The scaling assumes a uniform makeup throughout each of the vessels in the facility and ignores any behavior that may go on within a vessel, such as stratification and mixing. These aspects are considered in the Bottom-Up scaling which follows in the next section.

B.2.4.3 Bottom-Up Scaling

Bottom-Up scaling is a deductive approach that looks at specific phenomena and processes that may occur within a region. Local phenomena such as mixing, stored heat release, heat transfer and two-phase level are considered in this section.

B.2.5 Scaling of PANDA Facility

B.2.5.1 Facility Description and Test Characteristics

The PANDA facility is a full-vertical-scale multi-component integrated containment system test. The system scale is 1:25.

The facility is similar to the GIRAFFE facility in scope and is composed of the following regions:

- Reactor Vessel
- Drywell
- Wetwell
- GDCS pool
- 3 PCCs and associated pools
- IC and pool

The PANDA test will be used to provide steady-state separate effects data for the PCC, integral system performance and system interactions data for the containment, and concept demonstration of the startup and long-term operation of the PCCs. Details of the test facility and objectives are provided in Appendix A, Subsection A.3.1.3.

B.2.6 Integral Systems Tests Scaling Conclusions

Based on the findings from these scaling analyses, the following conclusions can be drawn about each of the test facilities:

- **GIST** – The GIST tests cover the period of late blowdown and GDCS initiation in a postulated LOCA event. The facility was scaled well to provide data for code qualification in the areas of GDCS initiation time and GDCS flow rate. The SBWR design changes since the time of the GIST test affect the data in such a way that GIST is not representative of the final SBWR design performance; however, nothing in the scaling precludes the use of GIST data for SBWR TRACG qualification.

- **GIRAFFE** – The GIRAFFE tests provide data on the long-term containment performance, PCC performance and systems interactions of the PCC and GDCS. The large heat losses in the Phase 1 tests result in deviation in the long-term containment performance. Since these heat losses can be modeled with high certainty with the system models, the data can still be used for TRACG qualification. The relatively small system scale results in rather large distortions in the Bottom-Up parameters. However, these local Bottom-Up effects are not expected to have a significant impact on the large scale system performance. The heat losses were substantially reduced in the Phase 2 configuration providing results more characteristic of the final SBWR design.

- **GIRAFFE Systems Interaction Tests (SIT)** - A scaling method was established for the existing GIRAFFE facility to simulate a LOCA from the late GDCS phase through early in the long-term containment phase. Preliminary results indicate that application of the scaling method described herein results in a facility able to provide data for this portion of an SBWR LOCA. In addition, the Bottom-Up distortions that result from a 1:400 scale facility have been quantified and the impacts discussed.

- **PANDA** – The PANDA facility is scaled very well and the data from this test can be used to qualify TRACG for long-term containment system and component performance as well as system interactions. The system is scaled to 1/25 of the final SBWR design for the time frame to be studied in the tests. The larger test scale results in reduced distortions in the Bottom-Up phenomena compared to GIRAFFE.

B.3 PANTHERS PCC/IC Component Test

The PANTHERS tests are full-scale component tests. Therefore, scaling analysis is not necessary for the majority of the facility. The facility includes a full-scale PCC unit (two modules) and one module of an IC unit. Complete descriptions of the PANTHERS facility and test objectives are contained in Appendix A, Subsection A.3.1.2.

The only two areas that are non-prototypical are the inlet for the IC test and the secondary side steam vent.

Although the IC test consists of one of the two modules that makeup an IC unit, the inlet line is the same size as for a full unit. The pressure drop and velocities will therefore be lower than they should be in the inlet. Because the inlet pressure drops are very low, the impact of this on the overall system will be small.

There are no significant scaling issues for the PANTHERS facility. The test will provide data for TRACG qualification of the PCC and IC performance. In addition, the tests will give information about scaling effects on PCC and IC heat transfer performance when compared to the smaller GIRAFFE and PANDA tests.

B.4 CRIEPI Natural Circulation Start-up Test

Tests were done at CRIEPI to study hydrodynamic instabilities during start up of a natural circulation BWR. The results of these tests and comparisons with TRACG are reported in several papers (References 64, 27 and 65). Figure B.4-1 shows the configuration of the test facility. The facility consists of two parallel heated channels feeding one chimney. The facility was scaled well to match the SBWR as described in Reference 64 and summarized below.

The basic equations of the drift-flux model were non-dimensionalized to arrive at the important non-dimensional numbers for hydrodynamic stability. The characteristic numbers are reported in Table B.4-1 for the SBWR and CRIEPI facility. The tests were not run at the full power conditions shown in the table. The full power conditions were selected to match N_{pch} and N_{sub} of the SBWR at full power. To arrive at the low power conditions, the power of the facility was ratioed down by the same amount as the SBWR for the desired conditions. The subcooling was set to match the SBWR also. A complete discussion of the method is described in Reference 64.

The comparison in the table is done for a representative case of 0.1 MPa system pressure. As shown in the table, the test facility compares very well with the SBWR. The most notable difference is in the flashing parameter, N_f. This difference is because the CRIEPI facility is about 70% as tall as the SBWR. The good general agreement in the important parameters of the SBWR and CRIEPI facility indicate that the results are applicable to the SBWR.

Tests were run at pressures of 0.1, 0.2, 0.35 and 0.5 MPa. The results are shown in Figure B.4-2. The figure shows the instability region in the heat flux-channel inlet subcooling plane as developed in Reference 64. Additionally, the expected SBWR conditions during start-up for these pressures are shown on the figure. The results indicate a significant amount of margin to unstable behavior in the SBWR. The margin increases as the pressure is increased.

Some additional tests in another facility showed unstable behavior over the entire range of conditions tested there (Reference 66). However, these tests were run at a much higher heat flux and subcooling than is representative of the SBWR. Figure B.4-2 shows that instability at higher heat flux and subcooling is consistent with the unstable region for the CRIEPI results.

Table B.4-1 Comparison of Non-Dimensional Parameters Between SBWR and CRIEPI

Nondimensional Parameters	Physical Meaning	Full Power Condition (7.2 mPa)		An example of startup condition (0.1 mPa)	
		Reactor	Test Facility	Reactor	Test Facility
Froude Number F_g	gravity to fluid inertia ratio	0.058	0.053	10.5×10^{-4}	7.6×10^{-4}
Loss coef. in channel ξ	pressure loss coefficient	3.4	2.7	6.9	5.7
Loss coef. at channel inlet $\xi_{c,in}$		50	30	50	30
Loss coef. at chimney exit (separator loss) $\xi_{r,ex}$		31	21	31	21
Phase change number N_{pch}	quantity of heating in channel	3.7	3.7	11.6	13.1
Subcool number N_{sub}	channel inlet subcooling	0.58	0.58	9.0	9.0
Flashing parameter N_f	quantity of flashing	0.057	0.036	67	46
Ratio of vapor density to liquid one R_{gal}	density ratio	0.052	0.052	6.2×10^{-4}	6.2×10^{-4}
inlet to that of dome pressure ρ_{a2}/ρ_{a1}		1.01	1.01	2.01	1.63
Nondimensional downcomer cross sectional area $A_{d,e}$	parameters depending on the test facility shape	1.05	1.11	1.05	1.11
Nondimensional chimney cross sectional area A_r		2.59	2.47	2.59	2.47
Nondimensional chimney length L_r		3.34	3.38	3.34	3.38
Nondimensional drift velocity v_{gj}	relative velocity between vapor phase and liquid phase	0.138	0.183	1.32	1.97
Arbitrary condensation parameter H_0	subcooled boiling	0.62	0.52	0.035	0.029
Peclet number P_E		120000	590000	13500	6000
Thermodynamic equilibrium quality at void departure point x_d		-0.047	-0.089	-3.45×10^{-4}	-3.81×10^{-4}

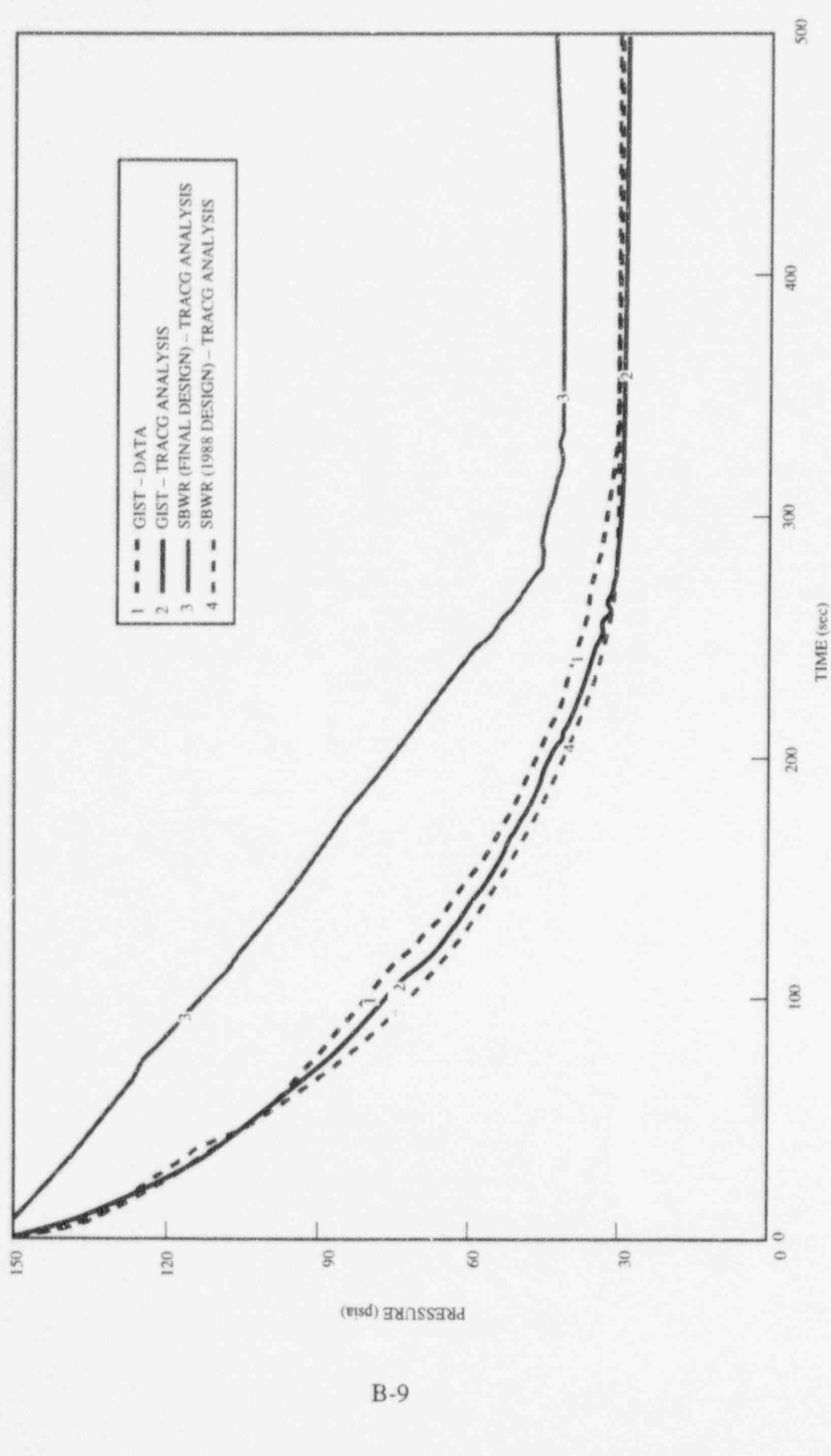

Figure B.2-1 Comparison of RPV Pressure Response for MSLB in GIST and SBWR

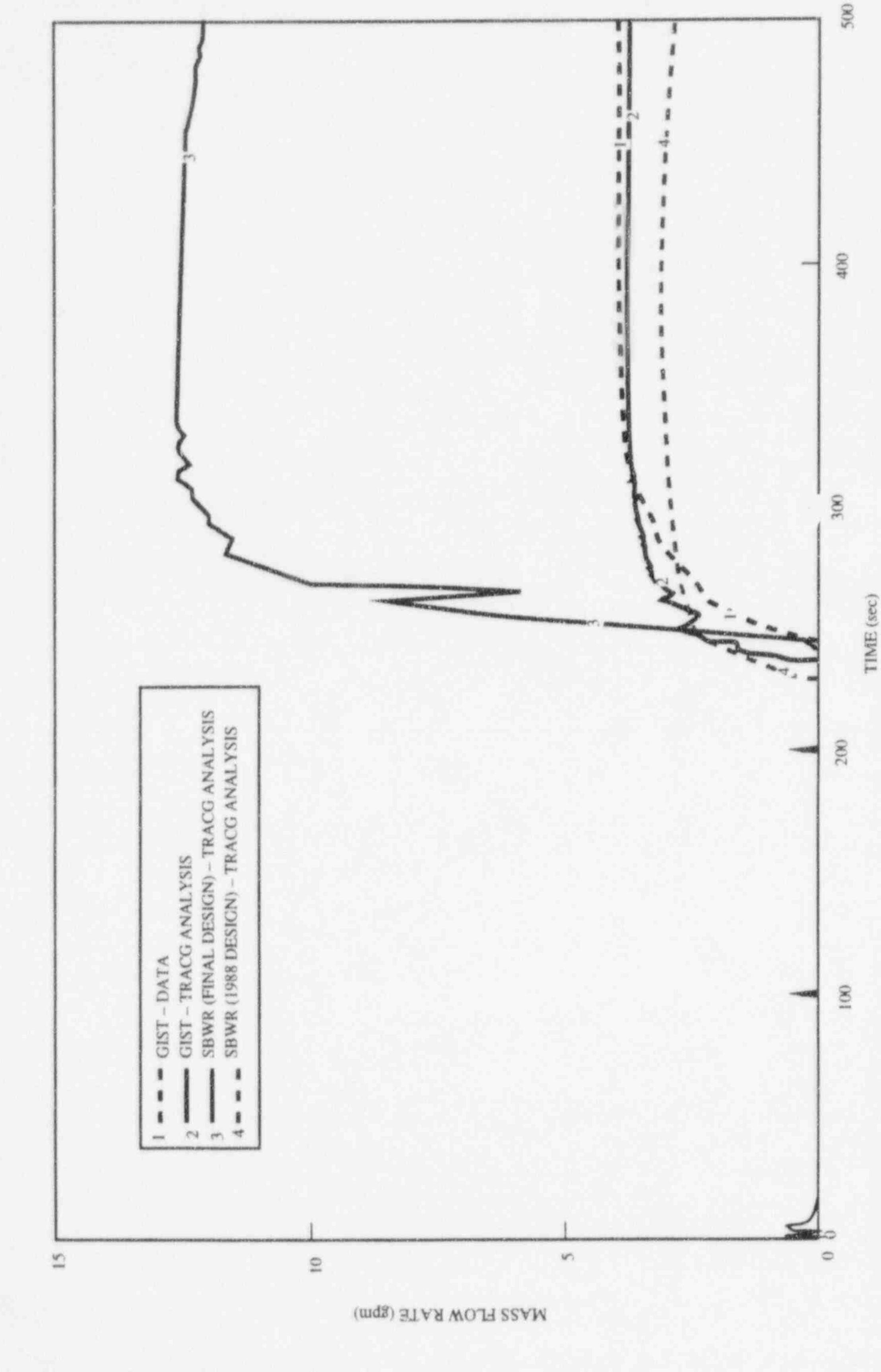

Figure B.2-2 Comparisons of GDCS Flow for MSLB in GIST and SBWR

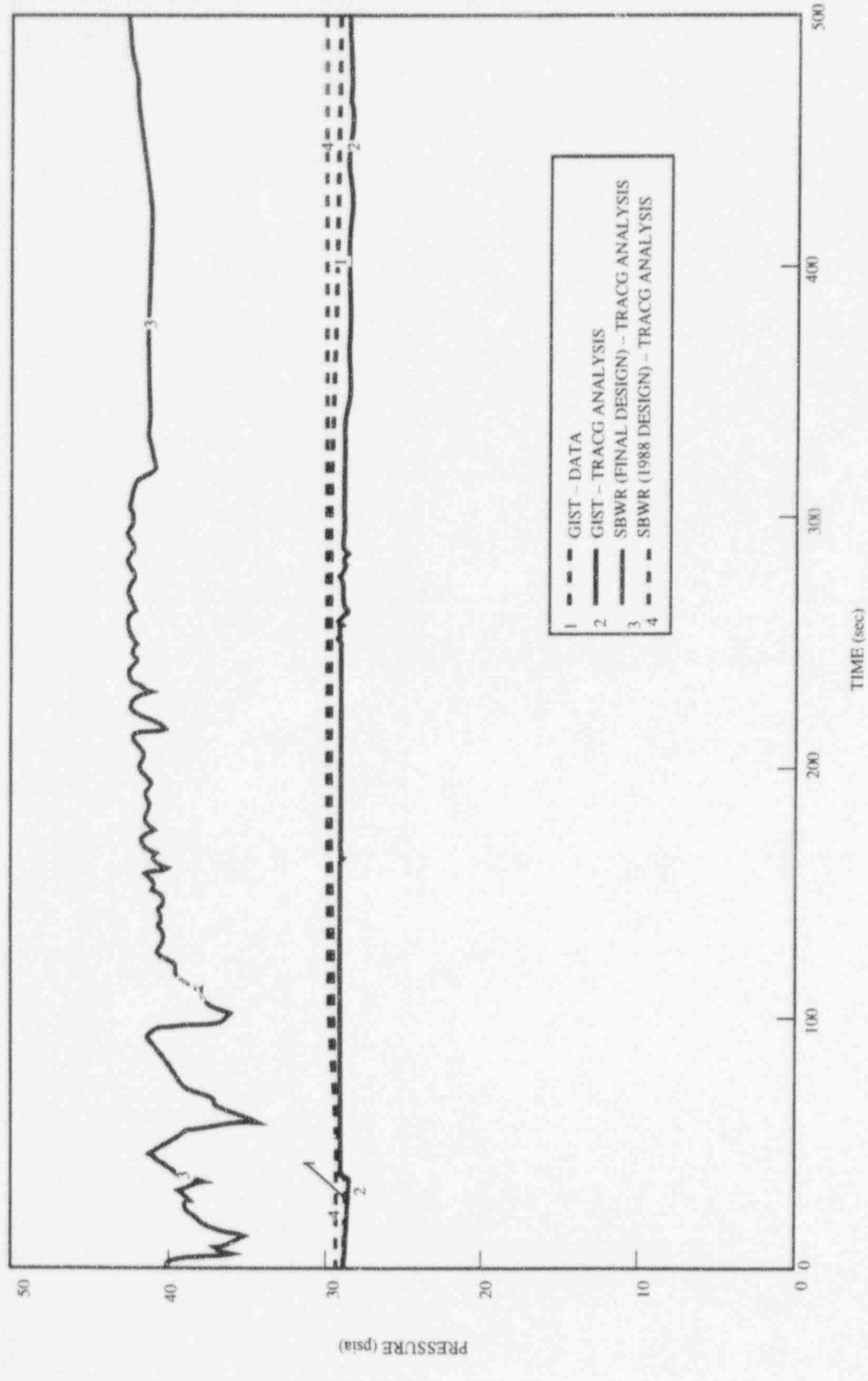

Figure B.2-3 Comparison of Drywell Pressure Response for MSLB in GIST and SBWR

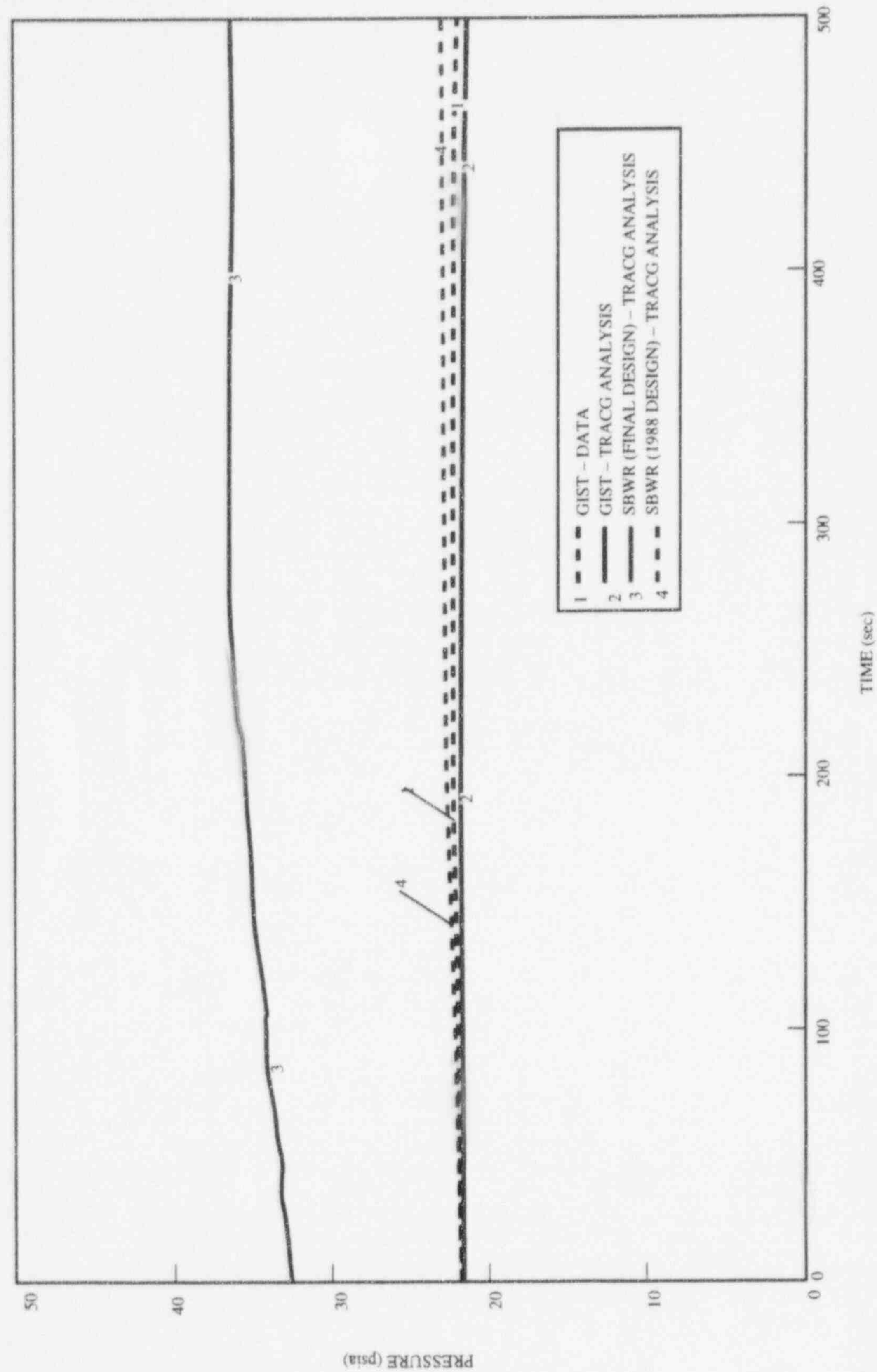

Figure B.2-4 Comparison of Wetwell Pressure Response for MSLB in GIST and SBWR

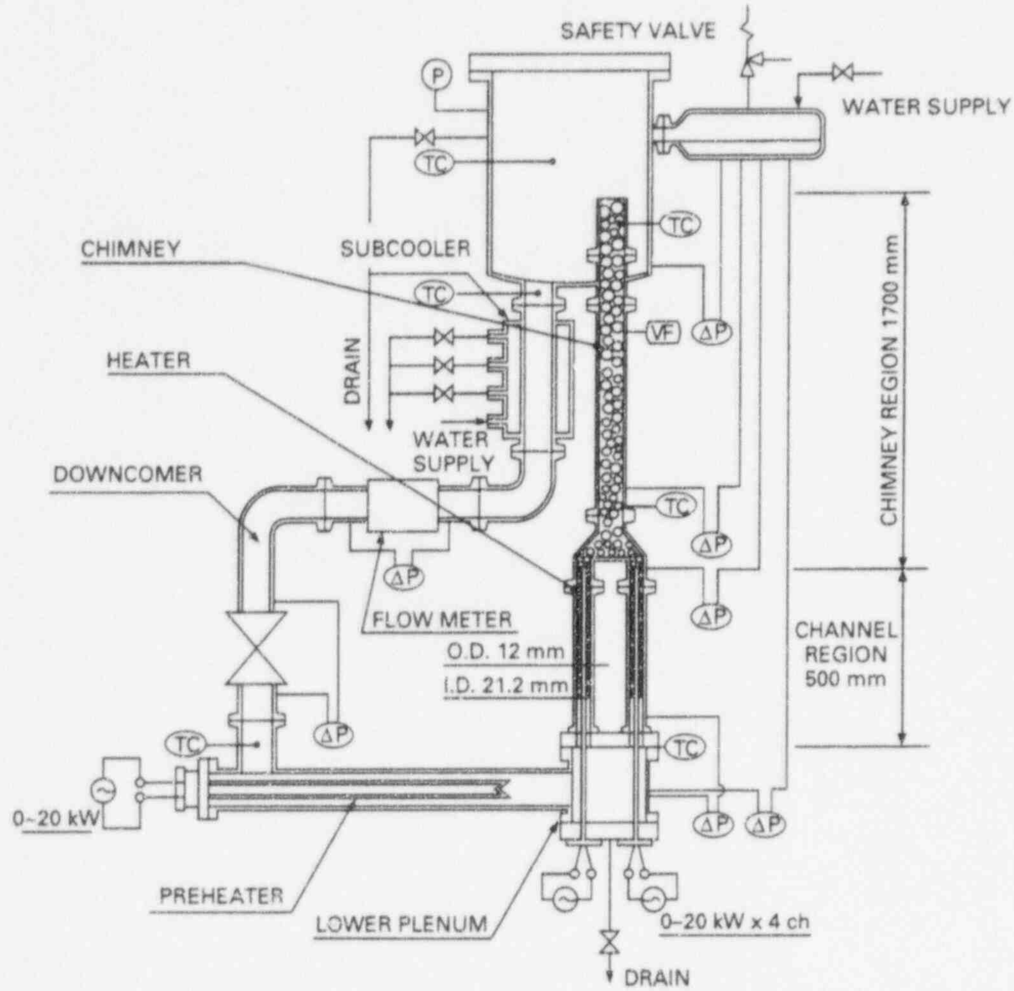

Figure B.4-1 CRIEPI Test Loop Outline

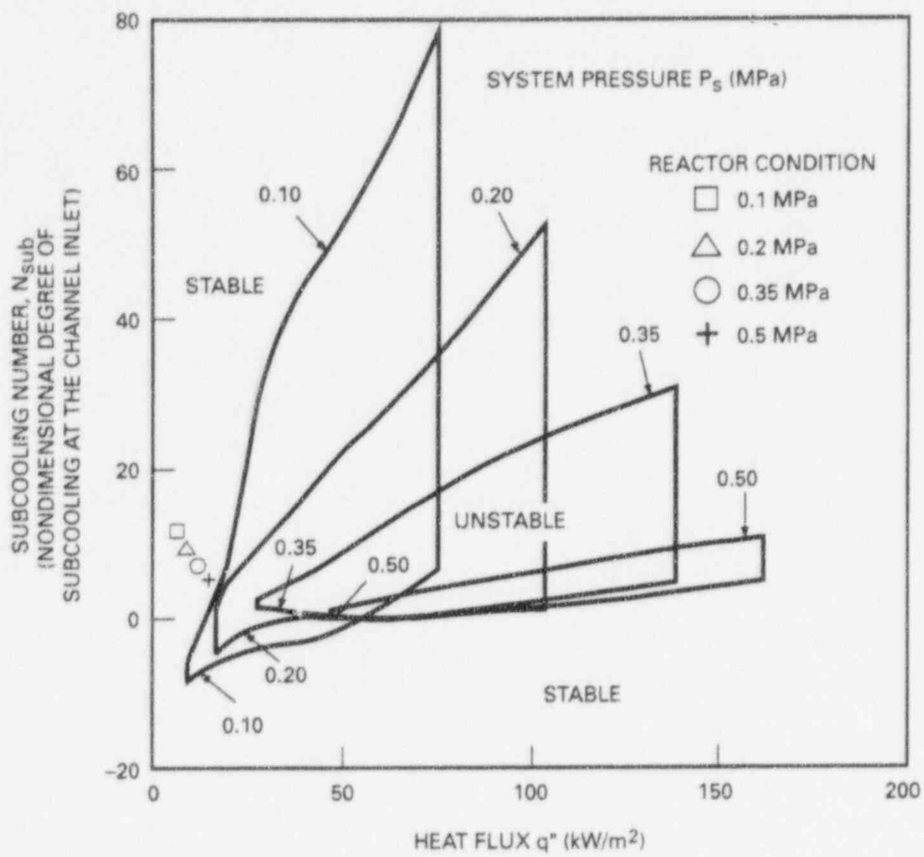

Figure B.4-2 CRIEPI Facility Stability Map Under Lower Pressure Startup Conditions and Representative Parameters for SBWR

ATTACHMENT B1 - DETAILED SCALING CALCULATIONS AND THEORY

Nomenclature and Abbreviations

A	Surface area [m^2]
a	Cross-sectional area [m^2]
c_p	Specific heat at constant pressure [J/kg K]
c_v	Specific heat at constant volume [J/kg K]
D	Diameter [m]
f	Friction factor
F	defined in text
F_n	defined in text
H	Height [m]
h	Specific enthalpy [J/kg]
h_{fg}	Latent heat of vaporization [J/kg]
g	Acceleration of gravity [9.81 m/s^2]
j, J	Volumetric flow rate [m^3/s]
k	ratio of specific heats = c_p/c_v
l	Length [m]
L	Sum of lengths [m]
M	Mass [kg]
$\dot{m}$	mass flow rate [ks/s]
p	Pressure [Pa]
$\wp$	perimeter [m]
$\dot{Q}$	Heat addition rate [W]
R	System Scale
T	Temperature [K]
t	Time [s]
u	Velocity [m/s]
V	Volume [m^3]
v	Specific volume [m^3/kg]
y	Mass fraction
z	Axial coordinate [m]

δ	Kronecker delta
υ	Viscosity
π	Non-dimensional number
ρ	Density [kg/m^3]
τ	Time constant [s]

Subscripts

G, g	Gas
L, ℓ	Liquid
LG	Change liquid to gas
R	Scaling factor between prototype and model
r	Reference

Additional subscripts are defined in the text.

Superscripts

o	Reference scale or variable
+	Non-dimesional variable

Abbreviations

DPV	Depressurization Valve
DW	Drywell
GDCS	Gravity-Driven Cooling System
GDLB	GDCS Line Break
H2TS	Hierarchical Two-Tier Scaling
IC	Isolation Condenser
ICS	Isolation Condenser System
LOCA	Loss-of-Coolant Accident
MSL	Main Steam Line
MSLB	Main Steam Line Break
PCC	Passive Containment Cooler
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
RPV	Reactor Pressure Vessel
SBWR	Simplified Boiling Water Reactor

SC	Pressure Suppression Chamber
SP	Suppression Pool
SRV	Safety/Relief Valve
WW	Wetwell

B1.1 Introduction

This Attachment contains the scaling equations and details of application of the scaling developed in Reference 32 to the four SBWR test facilities – PANTHERS, PANDA, GIST, and GIRAFFE. Sections B1.2 and B1.3 summarize the Bottom-Up and Top-Down equations used in scaling the facilities. Section B1.4 contains all of the calculated values for all of the tests. In addition, Section B1.4 discusses the application of the equations and the reference values used.

B1.2 Bottom-Up Scaling

B1.2.1 Introduction

When test facilities are scaled from a Top-Down perspective, many of the local, Bottom-Up phenomena will necessarily be distorted. It is generally not possible to scale without distortion both the Top-Down and Bottom-Up parameters in a reduced size facility. These local phenomena do not, however, significantly impact the overall system performance and, therefore, usefulness of the results. This section contains an expansion of the general Bottom-Up scaling contained in Section 3 of Reference 32.

B1.3 Top-Down Scaling

B1.3.1 Methodology

The general Top-Down scaling criteria for the SBWR are outlined in Section 2.4 of Reference 32. The resulting parameters are repeated here.

The six non-dimensional numbers are:

Enthalpy-pressure

$$\Pi_{hp} = \left\{ \frac{\Delta h^\circ}{\Delta p^\circ / \rho^\circ} \right\}$$

Phase-change

$$\Pi_{pch} = \left\{ \frac{\dot{Q}^{\circ}}{J^{\circ}\rho^{\circ}\Delta h^{\circ}} \right\}$$

Interfacial Phase Change

$$\Pi_{ipch} = \left\{ \frac{A_{LG}\dot{m}^{o}_{LG}}{J^{\circ}\rho^{\circ}} \right\}$$

Inertial Pressure Drop

$$\Pi_{in} = \left\{ \frac{\rho^{\circ}u^{\circ 2}_{r}}{\Delta p^{\circ}} \right\}$$

Submergence

$$\Pi_{sub} = \left\{ \frac{\rho_1 g H^{\circ}_{sub}}{\Delta p^{\circ}} \right\}$$

Hydrostatic Pressure Drop

$$\Pi_{hyd} = \left\{ \frac{\rho^{\circ} g L_g}{\Delta p^{\circ}} \right\}$$

Additionally, there are three time scales:

Volume time constant

$$\tau^{\circ} = \frac{V^{\circ}}{J^{\circ}}$$

Transit time constant

$$\tau_{tr} = \left\{ \frac{L_V}{u_r^o} \right\}; \qquad L_V = \Sigma \frac{a_n}{a_r} l_n$$

Inertial time constant

$$\tau_{in} = \left\{ \frac{L_I}{u_r^o} \right\}; \qquad L_I = \Sigma \frac{a_r}{a_n} l_n$$

and two geometric parameters:

Ratio of equivalent inertia and volume lengths

$$\frac{L_I}{L_V}$$

Total flow resistance

$$F = \Sigma F_n \frac{a_r^2}{a_n^2}; \qquad F_n = \frac{4 f_n l_n}{D_n} + k_n$$

As outlined in Reference 32, it is not necessary to preserve both Π_{in} and F; only their product,

$$\Pi_{loss} = \Pi_{in} * F,$$

must be preserved. Additionally, the time scales, τ_{tr} and τ_{in} will be small compared to τ^o and will, therefore, not affect the overall behavior of the system. Thus, it is sufficient to preserve τ^o as the dominant time-scale.

A brief discussion of the significance of each scaling parameter is given below. The first three Π value parameters are related to a volume with heat and mass entering or exiting, while the last three relate to flow in a pipe:

- *Enthalpy-pressure* relates additions of enthalpy to changes in the control volume pressure.
- *Phase-change* relates additions of heat to changes in fluid phase.

- *Interfacial Phase Change* essentially shows how well the phase change surface areas were modeled.
- *Inertial Pressure Drop* represents the pressure drop associated with the fluid velocity.
- *Submergence* represents the dynamic head needed to overcome the submergence of a pipe.
- *Hydrostatic Pressure Drop* indicates the pressure drop associated with fluid elevation changes.

Since the fluid properties are prototypic, the submergence and hydrostatic pressure drop numbers become a measure of how well the different elevations were maintained.

Additionally, two-phase behavior is important inside of the reactor vessel. The following list of two-phase parameters describes the scaling of this phenomena:

Void Fraction

$$\alpha$$

Volumetric flow ratio

$$\frac{J_g^\circ}{J^\circ}$$

Vaporization number

$$\frac{\dot{m}_{fg}^\circ V^\circ}{J^\circ \rho_g^\circ}$$

Pressure change time constant ratio

$$\frac{\tau_{prate}}{\tau^\circ}$$

Phase Change Number

$$\Pi_{pch} = \frac{Q^\circ}{J^\circ \rho^\circ h_{fg}^\circ}$$

Depressurization Number

$$\Pi_{dp} = \frac{Q^\circ J^\circ}{\Delta p^\circ V^\circ}$$

Flashing Number

$$\Pi_{fl} = \frac{\rho_L J_L^{\circ} \Delta h_{Axial}^{\circ}}{J^{\circ} \rho_g^{\circ} h_{fg}^{\circ}}$$

Density ratio

$$\frac{\rho_L}{\rho_g^{\circ}}$$

A review of the parameters shows that they will be scaled for a full height facility as long as the initial conditions of pressure, temperature, and mass fractions are preserved. The initial conditions are matched in all of the tests, therefore these parameters are not calculated specifically.

ATTACHMENT B2 - GDCS DRAINING

This attachment summarizes the development of the scaling used for draining the GDCS pool into the RPV. Figure B2-1 shows a schematic of the GDCS pool and RPV.

The one-dimensional momentum equation for a single phase fluid integrated over the length of the drain line is

$$P_2 - P_1 = -\rho L_I \frac{du_r}{dt} + \rho g L - \rho \frac{u_r^2}{2} F - \rho g H_{sub} \qquad \text{(B2-1)}$$

where

$$L_I = \Sigma \frac{a_r}{a_n} \ell_n$$

$$F = \Sigma F_n \left(\frac{a_r}{a_n} \right)^2$$

and

$$F_n = \frac{f_n l_n}{d_n} + k_n$$

ℓ_n, d_n, a_n, f_n, and k_n are the local length, diameter, area, friction, and form loss, respectively, in each section of pipe. a_r is a reference area.

Define the non-dimensional parameters,

$$u^+ = \frac{u_r}{u^\circ}$$

$$P^+ = \frac{P_2 - P_1}{P^\circ}$$

and

$$t^+ = \frac{t}{\tau}$$

where

$u^\circ = \sqrt{\dfrac{2gL}{F}}$ (the initial terminal velocity in the pipe)

$$P^\circ = \rho g L$$

$\tau = \dfrac{V^\circ}{J^\circ}$ (the time to drain the tank with flow J°)

$$J^\circ = a_r u^\circ$$

and

V° is the GDCS volume.

Substituting these into equation (B2-1) yields the non-dimensional momentum equation,

$$P^+ = -\frac{\Pi_{in}}{\Pi_f}\frac{du^+}{dt^+} + 1 - u^{+2} - \Pi_{sub}$$

where

$$\Pi_{in} = \frac{a_r}{V^\circ} L_I$$

$$\Pi_f = \frac{F}{2}$$

$$\Pi_{sub} = \frac{\rho g H_{sub}}{\rho g L}$$

When solved for the velocity this becomes,

$$u^{+2} = 1 - \frac{\Pi_{in}}{\Pi_f}\frac{du^+}{dt^+} - \Pi_{sub} - P^+ \qquad \text{(B2-2)}$$

To find the quasi-steady-state velocity in the pipe, set the fluid acceleration, du^+/dt^+, to zero. Also, let P^+ equal zero since both the RPV and GDCS are at about the same pressure for most of the time in which the GDCS is flowing. The quasi-steady-state velocity is then,

$$u^+ = (1 - \Pi_{sub})^{1/2} \qquad \text{(B2-3)}$$

To evaluate the inertial time constant for the fluid, look at the balance of the acceleration and hydrostatic terms in equation (B2-2),

$$\frac{u^+ - 0}{\Delta t^+} = \frac{\Pi_f}{\Pi_{in}}(1 - \Pi_{sub})$$

or

$$\Delta t^+ = \frac{\Pi_{in}}{\Pi_f}\frac{u^+}{1 - \Pi_{sub}}$$

From this,

$$\tau_{in} = \Delta t = \tau \Delta t^{+} = \tau \frac{\Pi_{in}}{\Pi_{f}} \left(1 - \Pi_{sub}\right)^{-1/2}$$

where the value of u^{+} has been substituted from equation (B2-3).

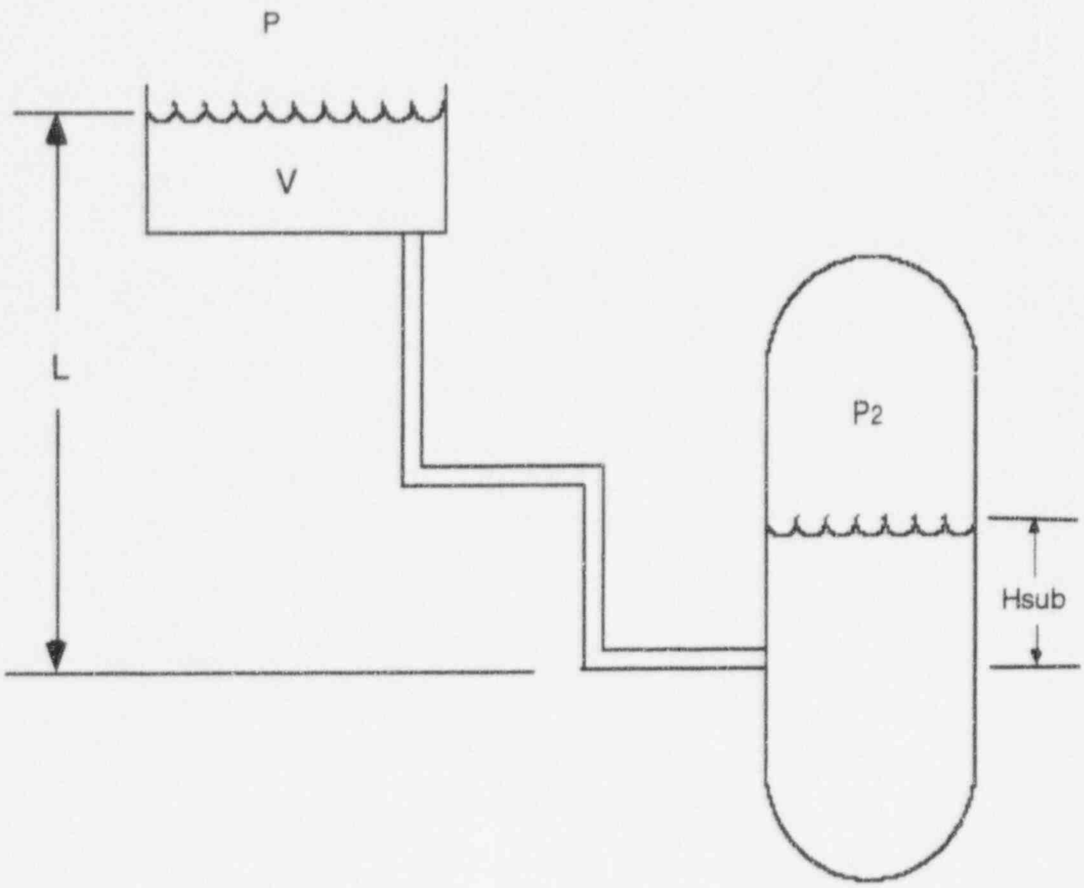

Figure B2-1 Schematic of Draining GDCS Pool

APPENDIX C - TRACG INTERACTION STUDIES

C.1 Introduction

If a LOCA were actually to occur in an SBWR, several of the limiting assumptions used in the licensing analysis may not (in fact, probably will not) apply. In particular, not all power may be lost, and non-safety grade systems and safety grade systems that are not engineered safety features (ESF) may be available to support accident management. This Appendix investigates interactions between active and non-ESF systems with the safety systems designed to operate during the LOCA, to determine if adverse effects due to interactions could result in conditions worse than the case if the non-ESF systems had not been available. The figure-of-merit used to measure the effect of system interactions inside the reactor vessel is the water level inside the chimney. Outside the vessel, the containment pressure and temperature are used. These studies are an extension of earlier work described in the SSAR which examined the effect of break location on the LOCA and the use of non-ESF systems to prevent core damage.

The TRACG code has been used for these studies. For interactions affecting the primary system response (inside the vessel) the TRACG input model for LOCA analysis was used. This input model provides a detailed representation of the reactor core, vessel internals and associated systems, but a less detailed representation of the containment. For interactions which may affect the containment response (outside the vessel) the TRACG input model used for containment response was used. This input model provides a more detailed representation of the containment and its systems, but a less detailed pressure vessel model. Both input models have been benchmarked to assure that they predict similar global response for the pressure vessel and containment.

Accident scenarios used for the study are similar to those used for LOCA licensing analysis, but additional systems are made available. The use of any additional systems is guided by the SBWR emergency procedure guidelines (EPGs).

C.2 Scenario Definition for Interaction Studies

The systems selected for the study were those that would likely be available and could produce adverse interactions with the ESF systems. Systems that would clearly benefit the system response were not considered. For example, with power and the feedwater system available, vessel inventory could be controlled and there would be no threat of core damage and no need for the passive systems. The Reactor Water Cleanup (RWCU) System is another beneficial system. It removes water from the vessel, cools it, and returns it through the feedwater line. For all but a feedwater line break, it provides heat removal capability in addition to the passive systems. The exception is for a feedwater line break, where operation of the RWCU System could reduce vessel inventory. This potentially adverse interaction is considered in the study.

For the several different breaks which were analyzed, three cases were considered:

- Loss of all AC power, except that provided from inverters
- On-site diesel generator power available
- Normal auxiliary power available

The first case is the basis used for the LOCA licensing analysis, and the results provide a measure of the system performance for the other cases where additional systems are available. The first case also provides an opportunity to examine system interactions between those safety systems that are expected to be available during the design basis accident. In all cases, the ESF systems were assumed to operate as designed.

C.3 Primary System Interaction Studies

The primary system interactions study investigated the effects of non-ESF systems on the vessel downcomer level and chimney level response. Several break locations were considered.

C.4 Containment Interaction Studies

The containment system interactions study investigated interactions between the ESF systems, and interactions of ESF systems with other systems which could be available for containment cooling without a loss of power.

C.5 Summary of Interaction Studies

The system interactions included in this study were those considered most likely to occur when some form of external power was available and which were not clearly beneficial to the operation of the ESF systems.