GE Nuclear Energy

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Licensing Topical Report

SBWR Test and Analysis Program Description

SBWR

GE Nuclear Energy

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Attention: Richard W. Borchardt, Director
Standardization Project Directorate

Subject: ***SBWR Test and Analysis Program Description,* NED0-32391 Revision A**

This letter transmits Revision A of the *SBWR Test and Analysis Program Description* report, NED0-32391, for your review (Attachment 1). The report provides a comprehensive, integrated plan that addresses the testing and analysis elements needed for analysis of the SBWR performance. In particular, this document describes the final Test Plan (Appendix A).

Sincerely,

P. W. Marriott, Manager
Advanced Plant Technologies

Enclosure: *SBWR Test and Analysis Program Description (TAPD),* NED0-32391, Revision A

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SBWR Test and Analysis Program Description

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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
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
CRIEPI	Central Research Institute of Electric Power Industry
CSAU	Code Scaling, Applicability and Uncertainty
CSHT	Core Spray Heat Transfer
DBA	Design Basis Accident
DC	Direct Current
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 Accident (LOCA), Anticipated Transient 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 the 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. Analysis has 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.

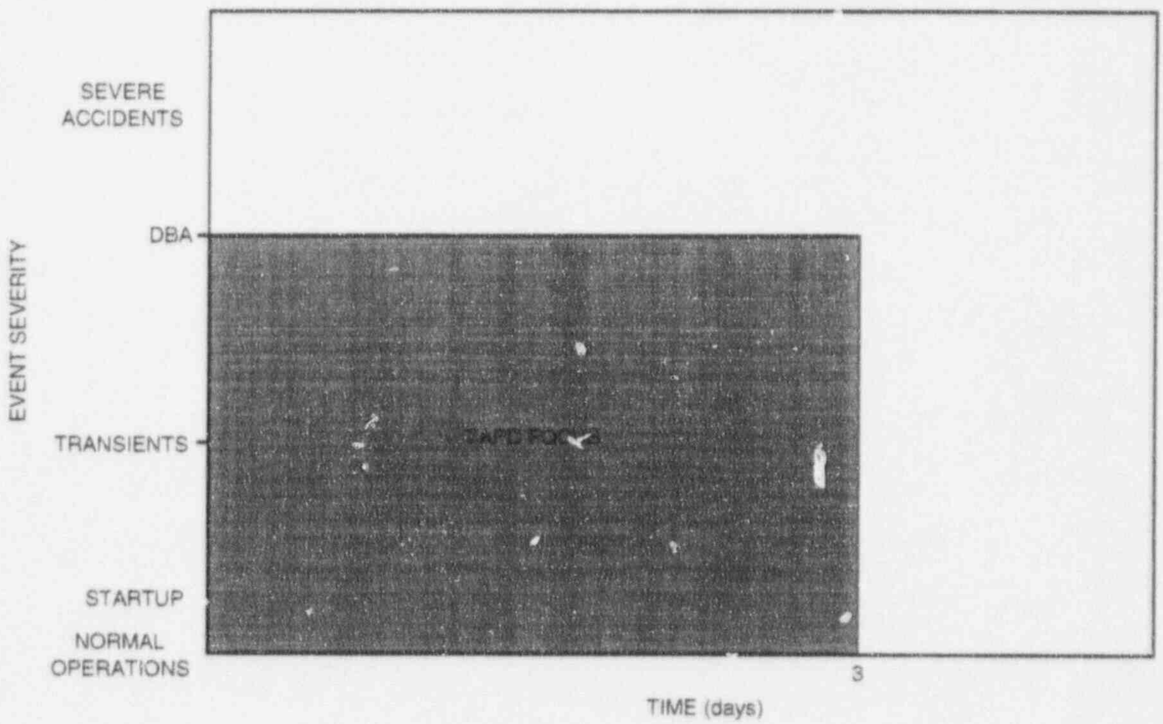


Figure 1.1-1. TAPD Focus

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. GE has chosen to perform SSAR analyses using the Best Estimate plus uncertainty method [10CFR50.46(a)(1)(i)]. The Level 2 application of TRACG includes LOCA analyses, transients, ATWS and Stability Analyses for the reactor and containment.

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 (Δ CPR) 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 (Section 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 over a hundred specific governing phenomena (summarized in Table 1.2-3 of this report), of which over half 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 21.

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 noncondensables, 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.

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.

1.2.2.1.3 PANDA

PANDA is an experimental program to be run by the Paul Scherrer Institut of 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 noncondensables 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 has 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 of this report 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 FMCRDs
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 1.2-3. TRACG SBWR Qualification

Region	Phenomena Identified	Major Phenomena
Lower Plenum	9	4
Bypass	10	6
Core	39	29
Guide Tube	7	3
Downcomer	7	6
Upper Plenum	5	4
Steam Separator	3	3
Steam Dryer	2	0
Steam Dome	3	0
Steamline	6	5
Containment	68	48
Total	**159**	**108**

NOTE:

Key SBWR Phenomena Identified (summary)

- Phenomena are ranked according to importance and major phenomena identified.

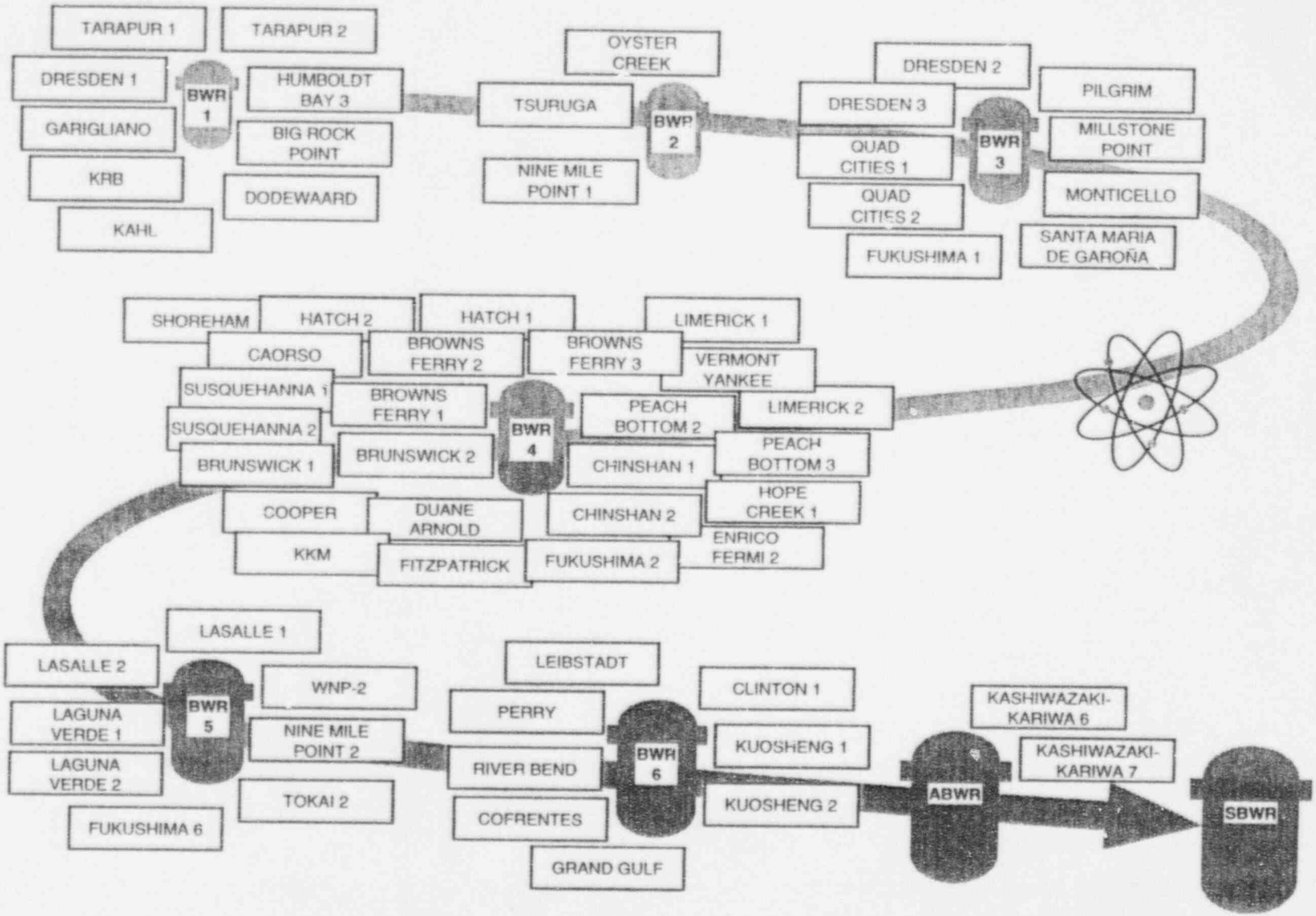


Figure 1.2-1. Evolution of the BWR

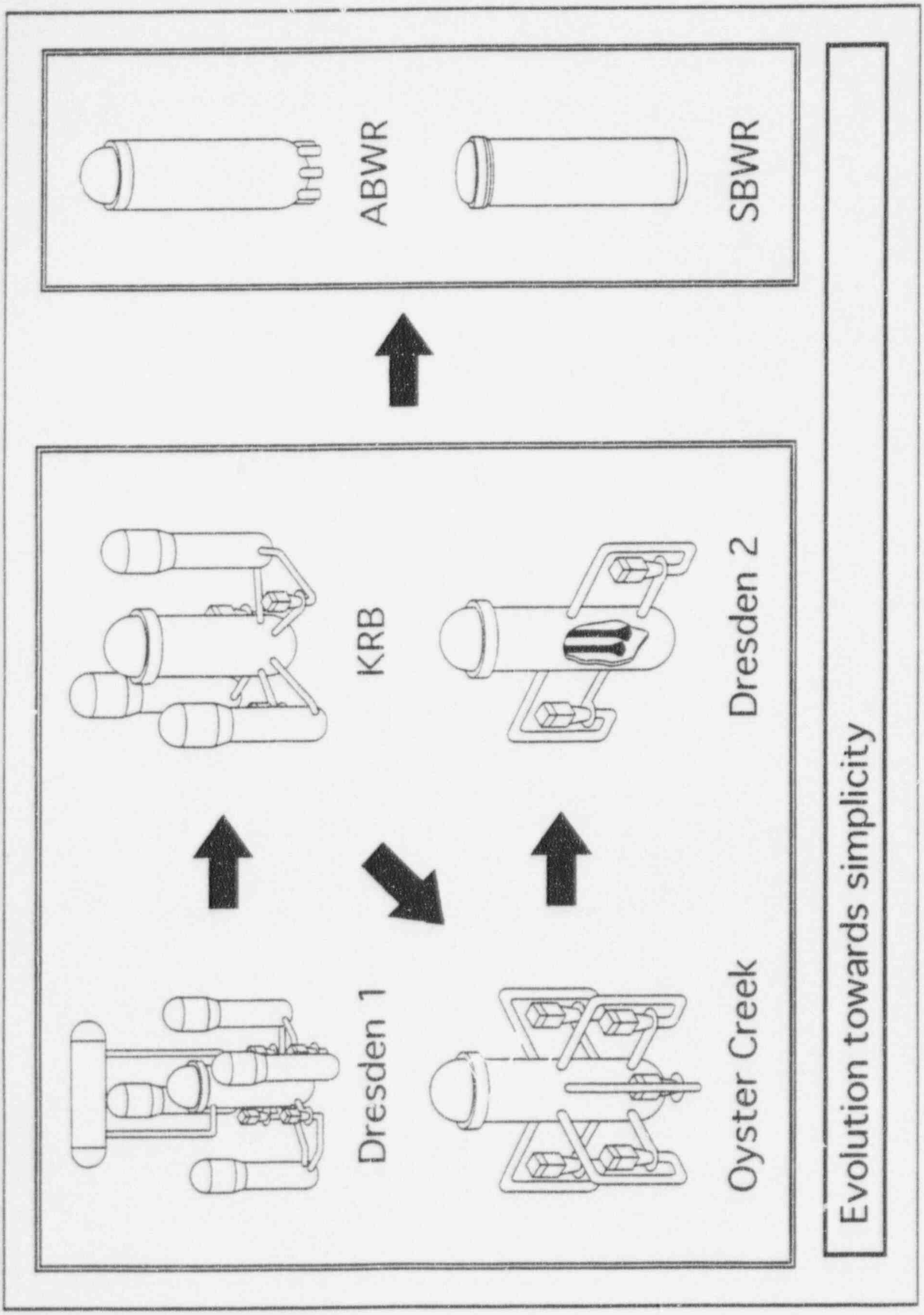
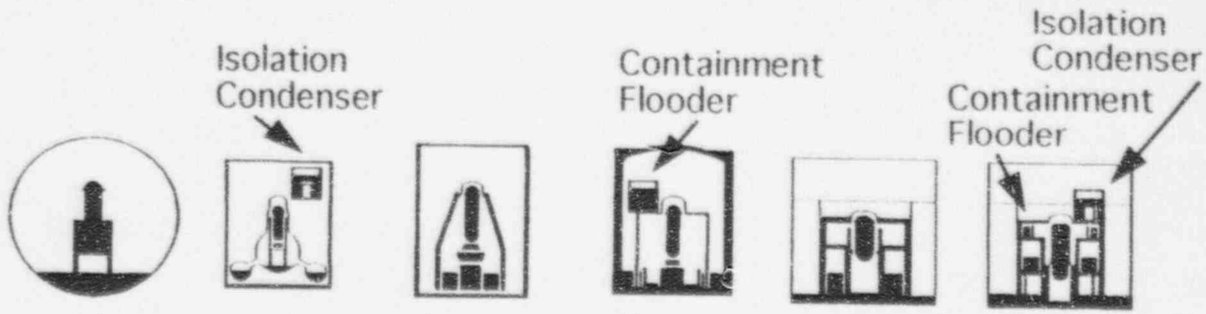


Figure 1.2-2. Evolution of the BWR



	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 x 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 of 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. For the phenomena generic to all BWRs, the evaluation consisted of confirming that data exist in the BWR database covering the phenomena. For the SBWR-unique phenomena, a more formal evaluation process was implemented. 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 component and 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.

This process is illustrated with the help of an example. One of the phenomena considered important for the LOCA/ECCS events is critical flow from the downcomer region, including the effects of break uncovery and two-phase break flow. This is listed as Phenomenon E1 in Table 2.3-1, "SBWR PIRT for LOCA/ECCS". Because it is given an importance ranking of 7 for the large break (blowdown period), it is a candidate for incorporation in the composite table of highly ranked phenomena in Section 4. This issue is also listed in the Bottom-Up process, Table 3.2-1, "SBWR Thermal-Hydraulic Phenomena" under B21, Nuclear Boiler System. It is evaluated in Table 3.3-1, "Issue Evaluations Summary" (#13) as having a sufficient data base. It is carried forward to Table 4.1-1a, "Composite List of Highly Ranked Phenomena for LOCA/ECCS" (Item E1). Section 5 shows that the phenomena of critical flow, including break uncovery and a range of upstream two-phase conditions (E1), are covered by the PSTF vessel blowdown, Edwards depressurization test and the Marviken Critical Flow tests in Table 5.1-1, "Separate Effects Tests for TRACG Qualification for SBWR - Reactor Vessel and Core". The integral system tests which cover the phenomena are found in Table 5.3-1 (TLTA and FIST large break tests and the SSTF test). Table 5.5-1 shows that Item E1 is covered by Separate Effects, Component and Integral System Tests. Because of this, it is not included in Table 6.1-1, which lists the additional needs for TRACG qualification.

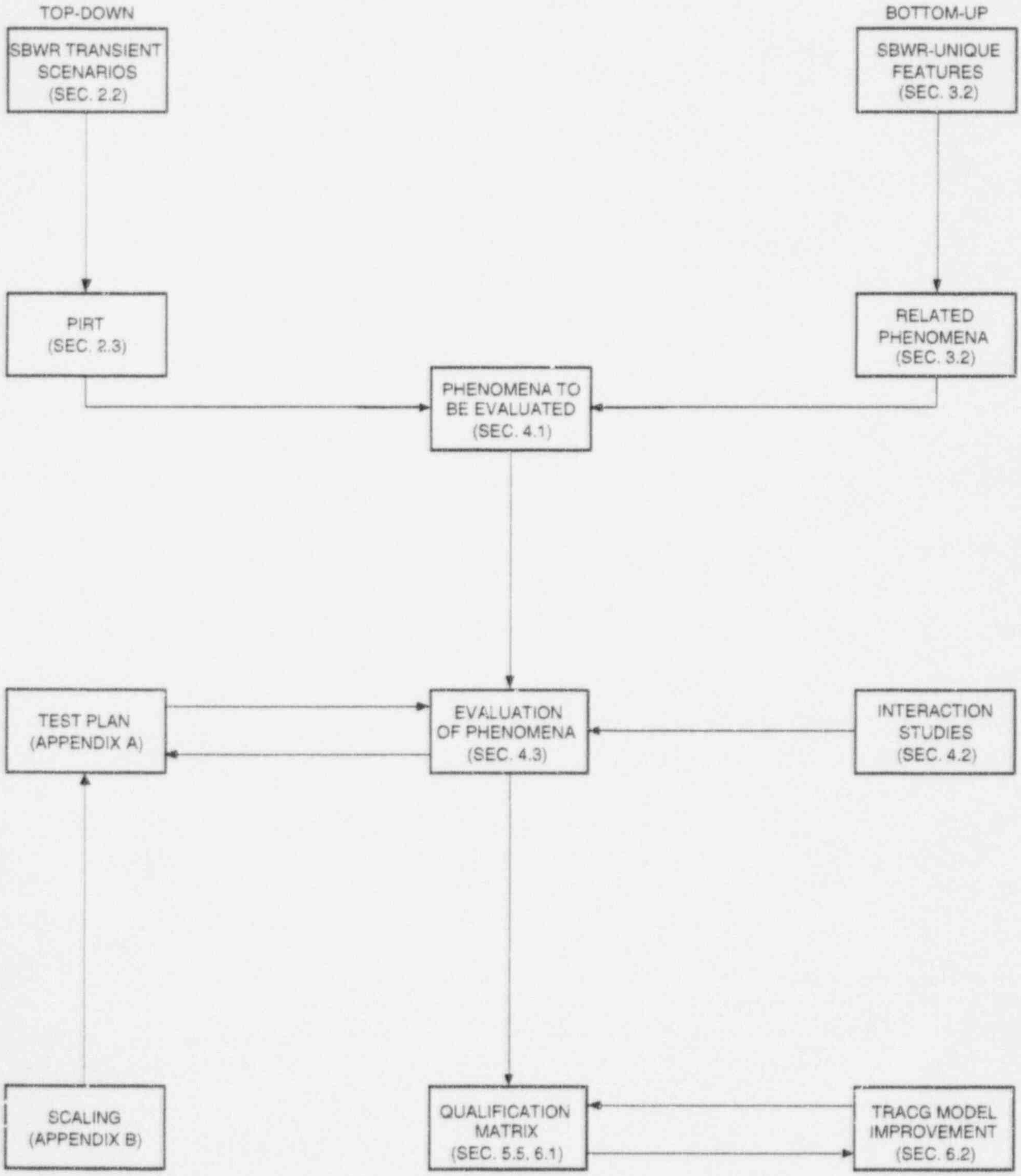


Figure 1.3-1. Strategy for Determination of Test Needs

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 Chapters 6 and 15 will be re-performed with the final version of the TRACG Code.

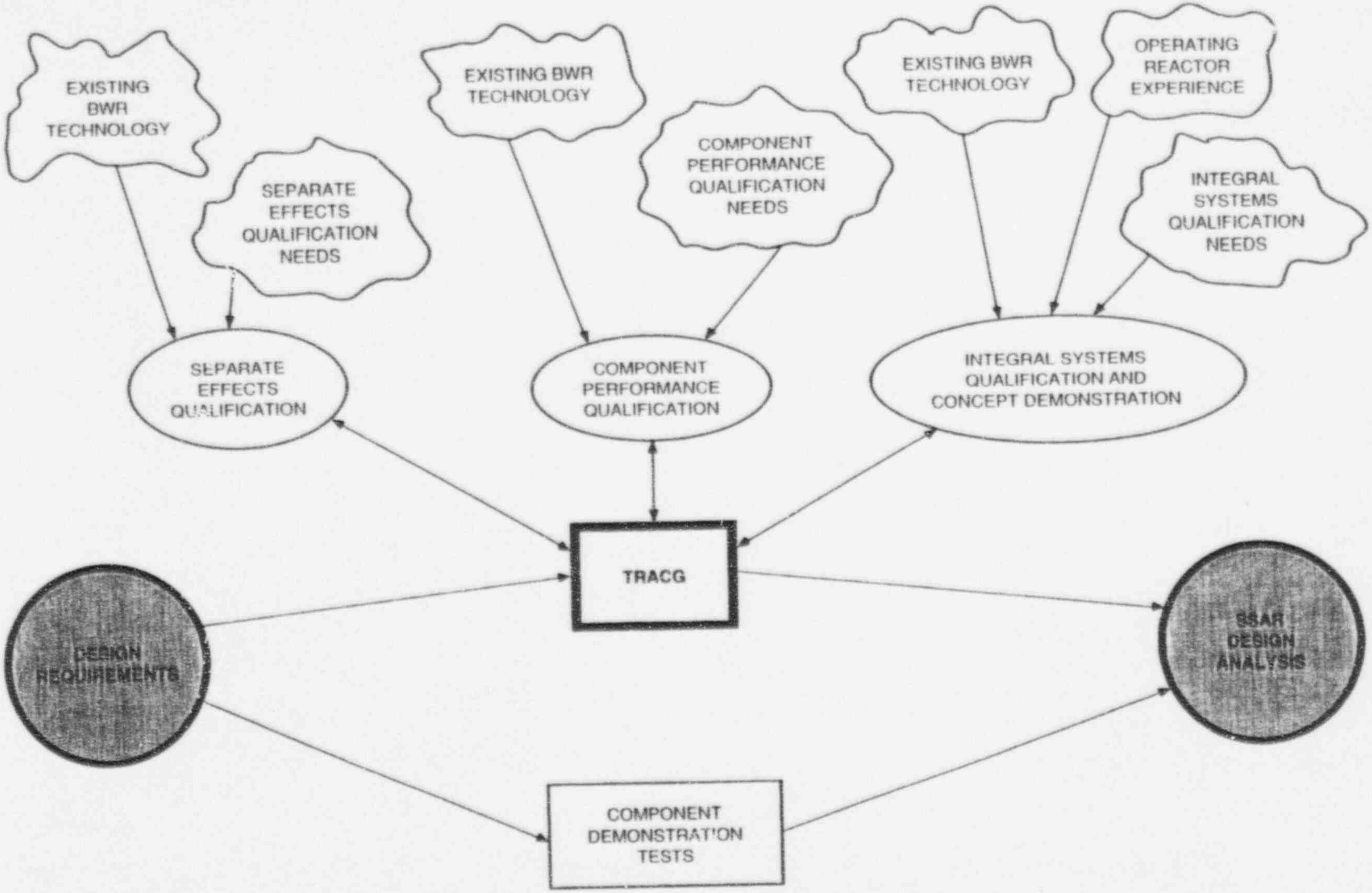


Figure 1.4-1. Technology Basis for SBWR Design

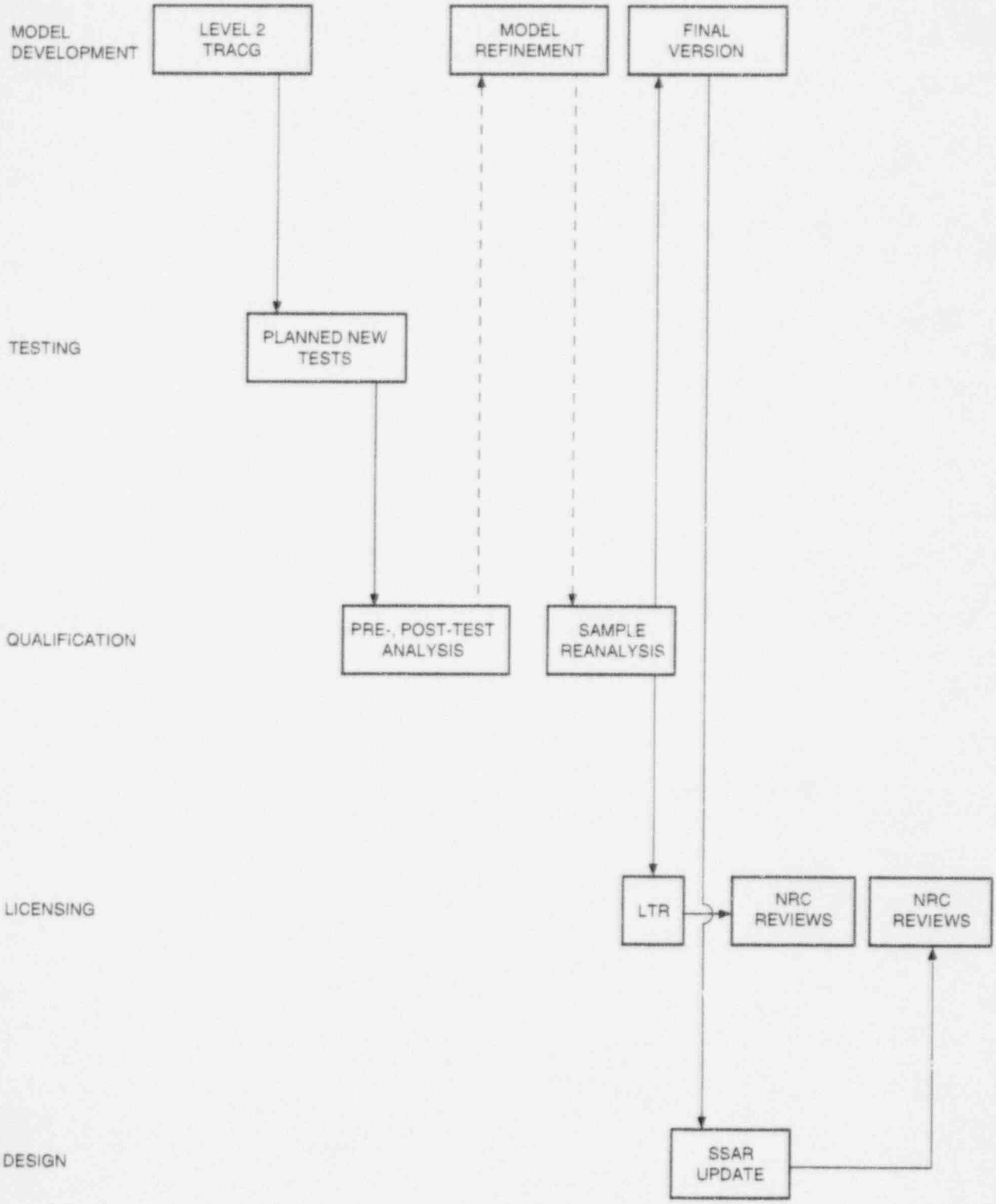


Figure 1.4-2. Overall Test and Analysis Plan

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). Section 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). For a brief view of the output of this process, see the first sheet of Table 2.3-1. 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 of 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. 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 pressurize the drywell, clearing the main containment vents and the PCCS vents. First, noncondensable gas and then steam flows through the vents and into the suppression pool. The steam is condensed in the pool and the noncondensable 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 noncondensable 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 noncondensable 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 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 noncondensable 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. The increase in drywell pressure establishes flow through the PCCS, which also picks up part of the blowdown energy. For the GDCS break, this period of the accident lasts less than 10 minutes.
- ***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 noncondensable 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 reestablished through the PCCS. The PCCS heat removal capacity, even while recycling noncondensable gas back to the wetwell, is sufficient to handle the steam generated by decay heat, and the main vents are not reopened. This period of the accident is expected to last approximately 3 hours for the GDCS 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 break is below the drywell pressure for the large steam line break. During the 72 hours which define 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 till 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 figure 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. 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.

2.2.2 Anticipated Transients

As with the LOCA, anticipated transients are discussed in the SSAR (Chapter 15) and no additional analyses are 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 steamlines. 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 steamlines.

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.

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 2.2-5). 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 In Steady-State Operation

In *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 2.2-5). 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. In Figure 2.2-6, 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 show that there is large margin for stability. This indicates that the SBWR is very stable under normal operation conditions.

2.2.4.2 Of the 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 2.2-5. In Figure 2.2-6, both of these transient events are seen to result in power/flow conditions that are well below the exclusion region.

2.2.4.3 Under ATWS Conditions

Under *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. The injection of boron will eventually shut down the reactor and mitigate the situation.

2.2.4.4 During 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 rpv < 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:

(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.

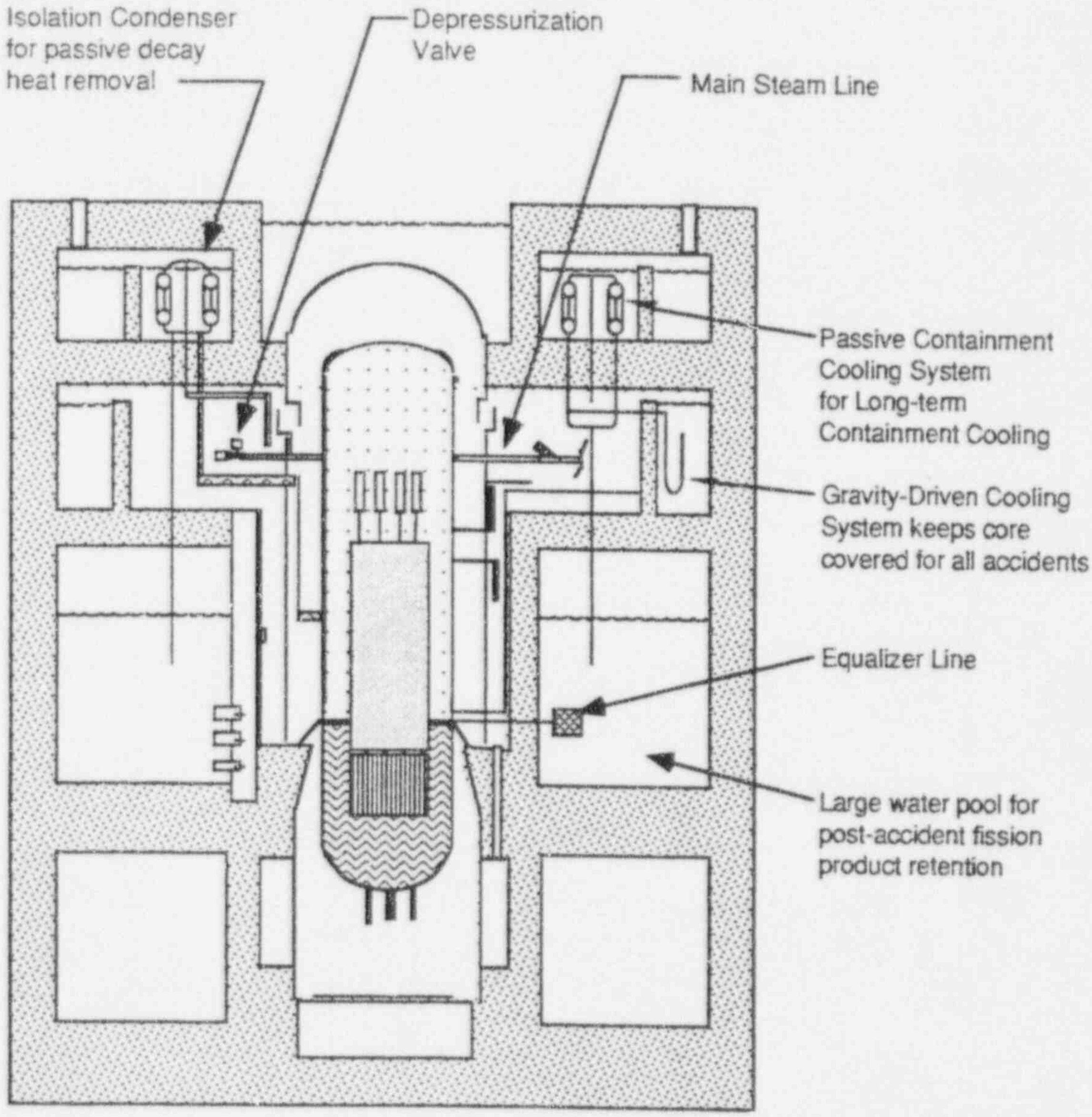


Figure 2.2-1. SBWR Passive Safety Systems

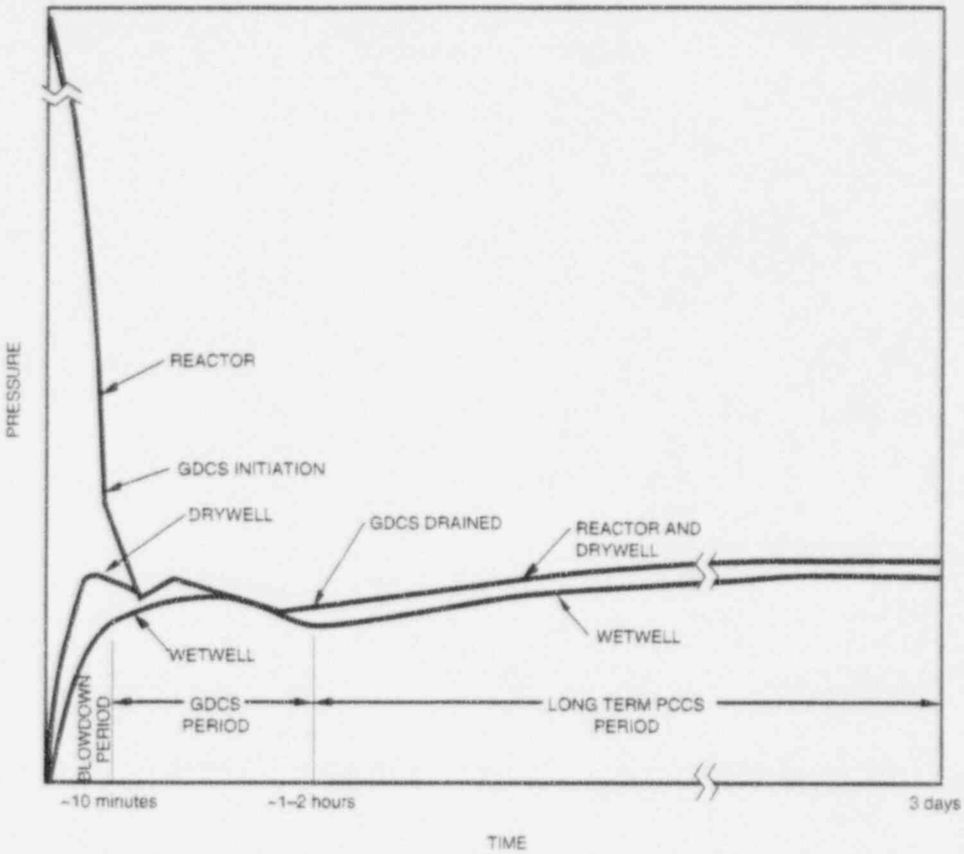


Figure 2.2-2. Phases of the LOCA Transient

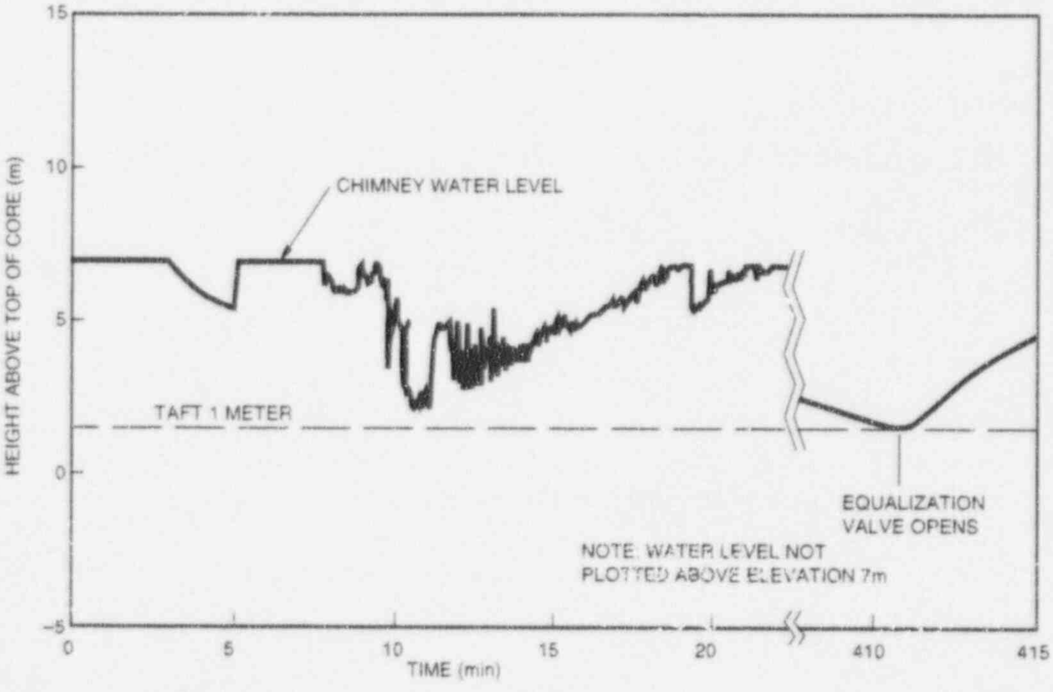


Figure 2.2-3. GDCS Line Break Reactor Water Level vs. Time

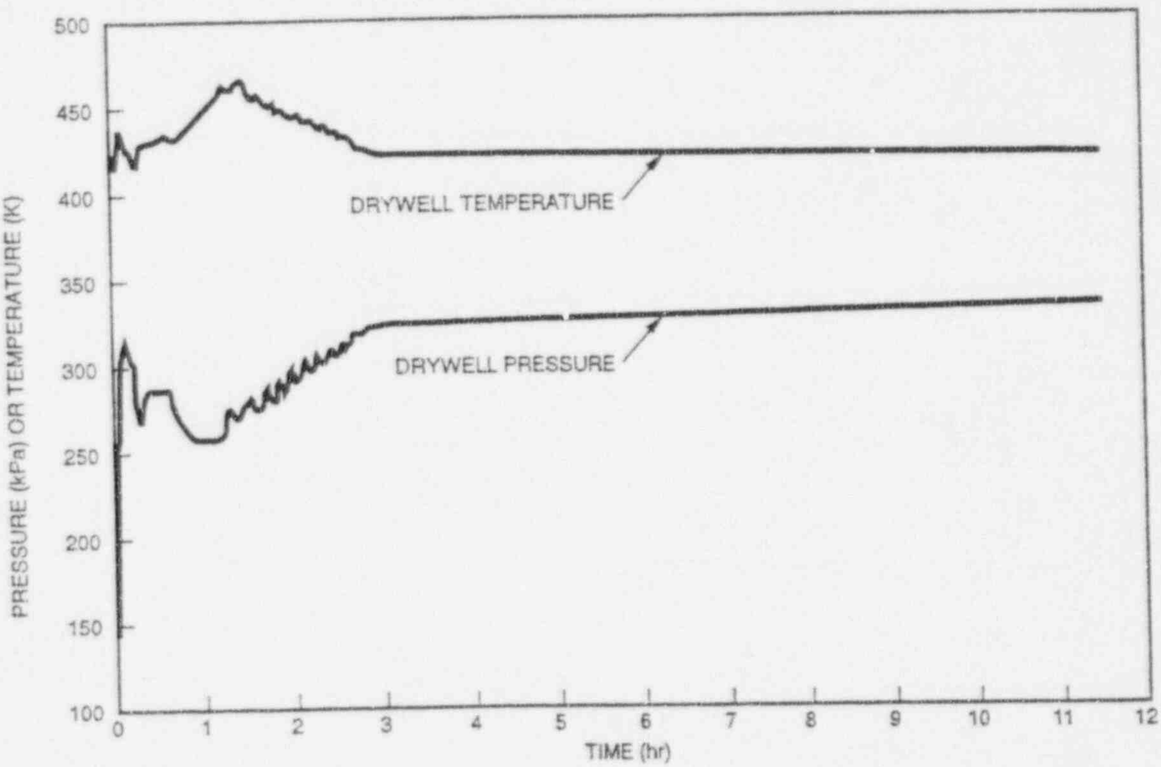


Figure 2.2-4. GDCS Line Break Containment Pressure and Temperature vs. Time

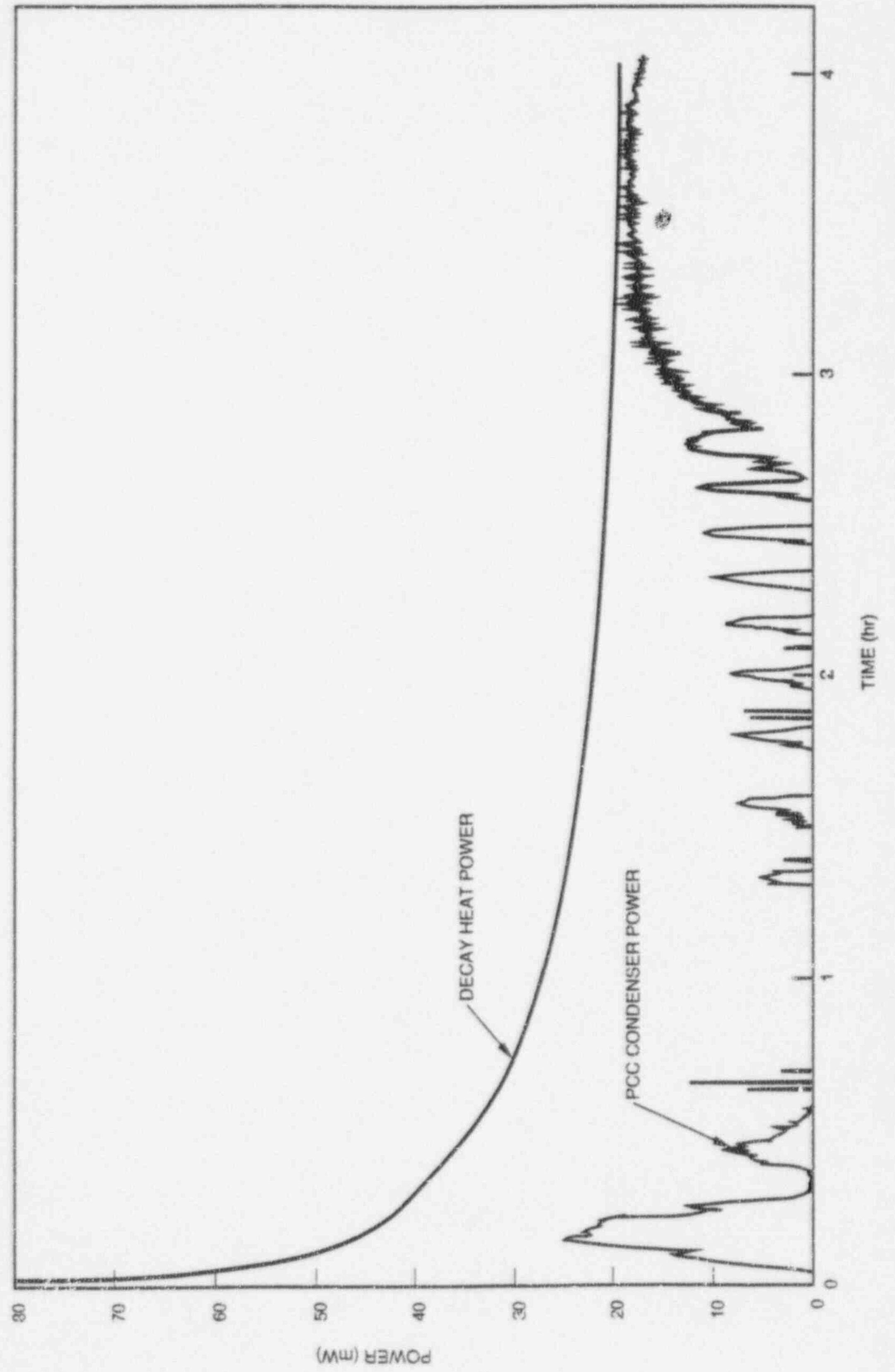


Figure 2.2-5. GDCS Line Break Decay Heat and PCC Power vs. Time

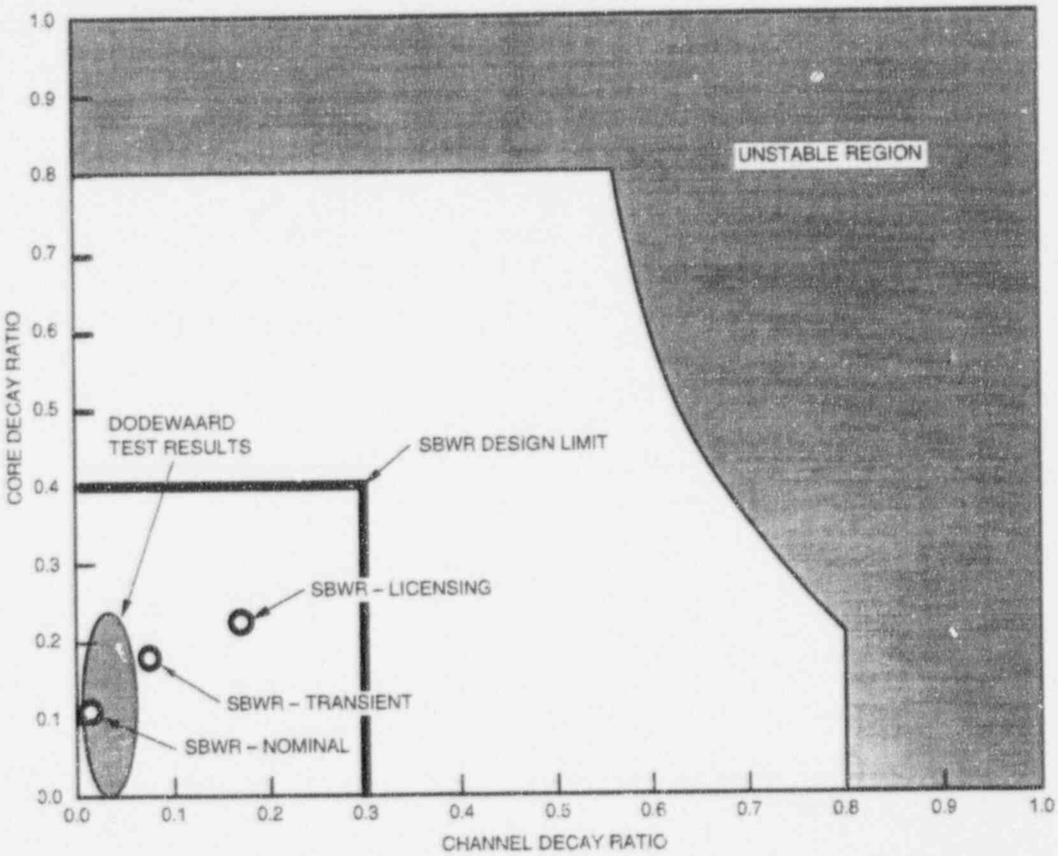


Figure 2.2-6. SBWR Stability Design Criteria and Performance

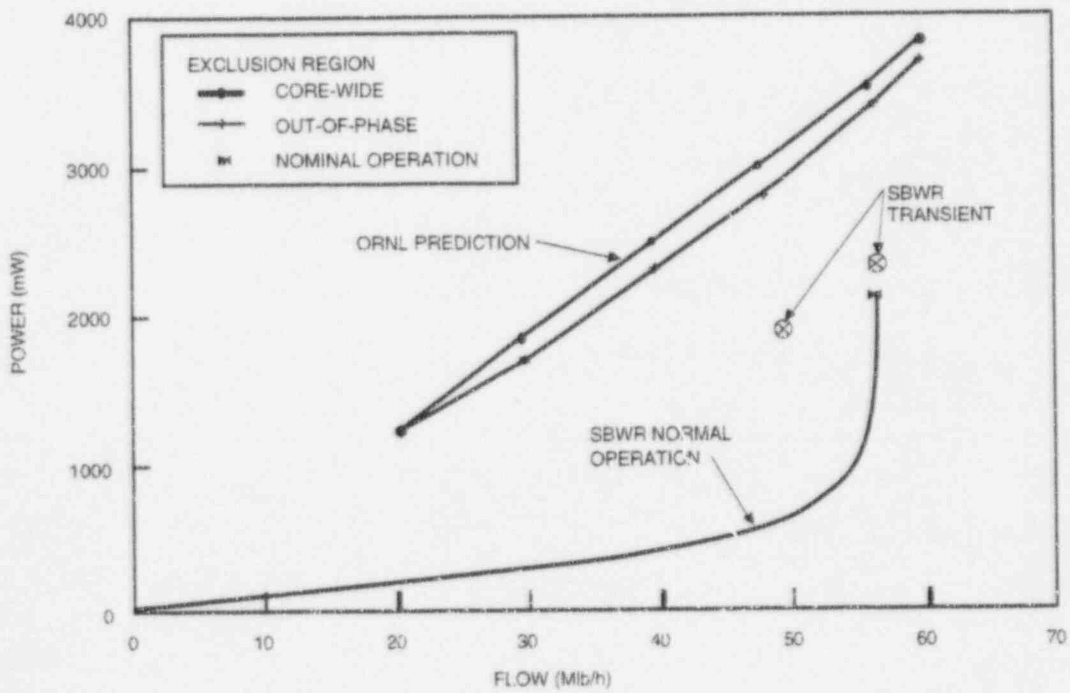


Figure 2.2-7. SBWR Power/Flow Map Comparison With Calculated Stability Limit

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. Tables were developed for small break LOCAs, large break LOCAs, operational transients, stability and reactivity insertion accidents such as control rod withdrawal error and control rod drop. In each case, the importance of the phenomena was evaluated for each reactor region: lower plenum, core, upper plenum/chimney, downcomer, containment, etc. For the LOCA events, the tables were further subdivided into the blowdown, GDCS and long-term periods of the transients.

The tables were first condensed to include only the highly ranked phenomena. 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. The phenomena that are unique to SBWR are given primary emphasis in 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 two purposes. First, the capabilities of the TRACG models are examined to see if all the important phenomena can be treated. Secondly, the qualification database is examined for completeness against the important phenomena. The second task is discussed in Chapter 5.

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. These periods are described in previous sections and shown in Figure 2.2-1. 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.).

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 Chapter 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 Section 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

In order to perform a systematic and thorough review of all systems in the design, the SBWR System Structure (Figure 3.2-1) was used as a starting point.

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 and Tables 2.3-1 and 2.3-2, to determine any new issues associated with that unique feature. For each of the issues, associated important thermal-hydraulic phenomena were identified.

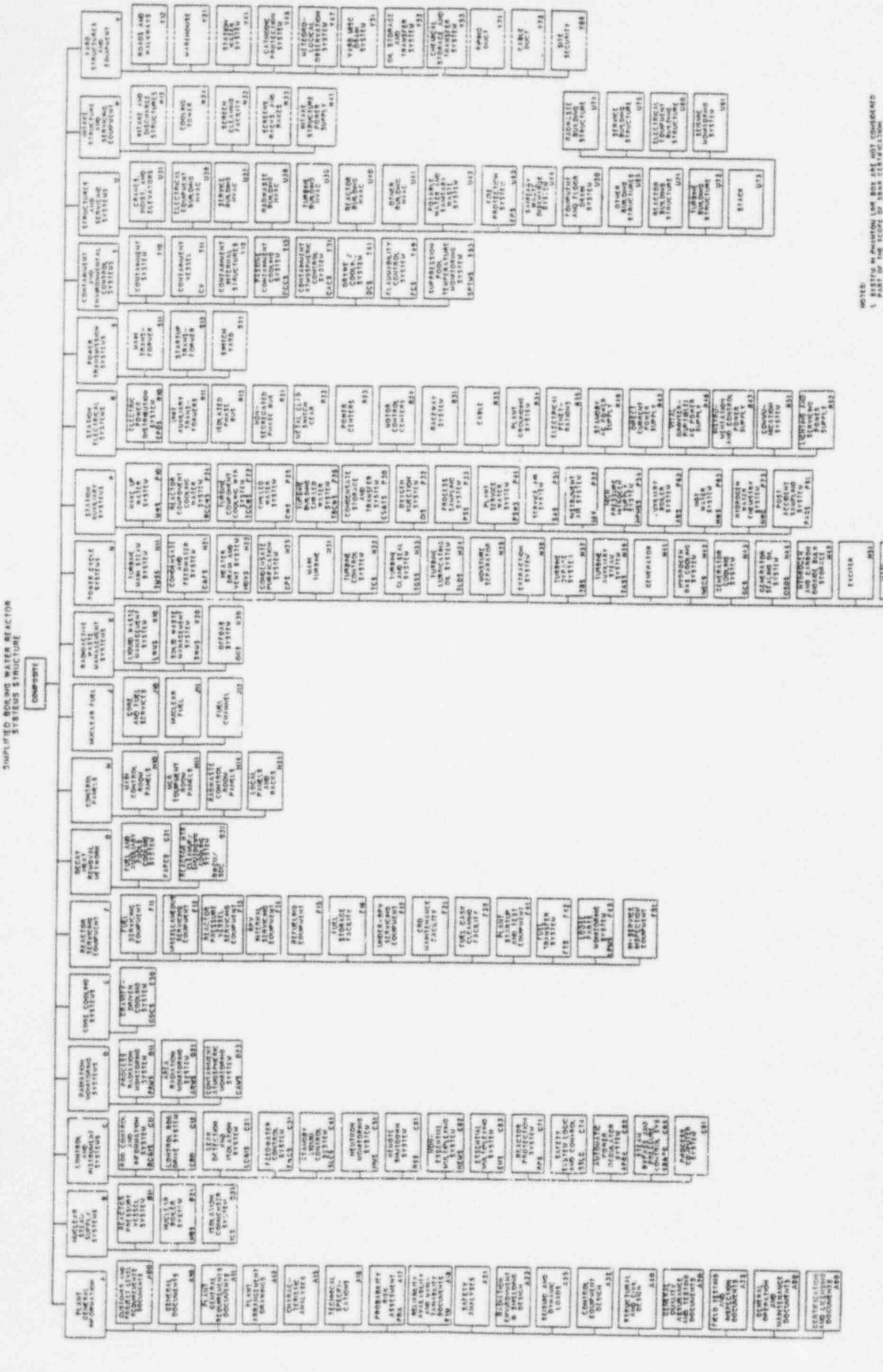


Figure 3.2-1. SBWR Product Structure

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 PCCS.

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 to investigate the effect of both active and passive systems which could be available to support Engineered Systems Feature (ESF) systems during a LOCA; and 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 [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 [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 have included those which were considered the most likely to occur when some form of external 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 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 Sections 4.3.1 and 4.3.2 for LOCA and Transients, respectively. Section 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 the SRV discharge to the suppression pool (critical flow, pool stratfication and heatup, etc.).

5.0 MATRIX OF TESTS NEEDED FOR SBWR PERFORMANCE ANALYSIS

The 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 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 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 can be found in the Test Plan in Appendix A. The Analysis Plan in Appendix A shows specifically which tests will be used for blind predictions, and which tests will be used for post-test analysis.

6.2 Use of Data for TRACG Model Improvement and Validation

The TRACG computer code has been 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 controlled Level 2 status under the GE-NE quality assurance procedures for computer codes. If changes in the nodalization are indicated, calculations affected by the changes will be redone and reverified.

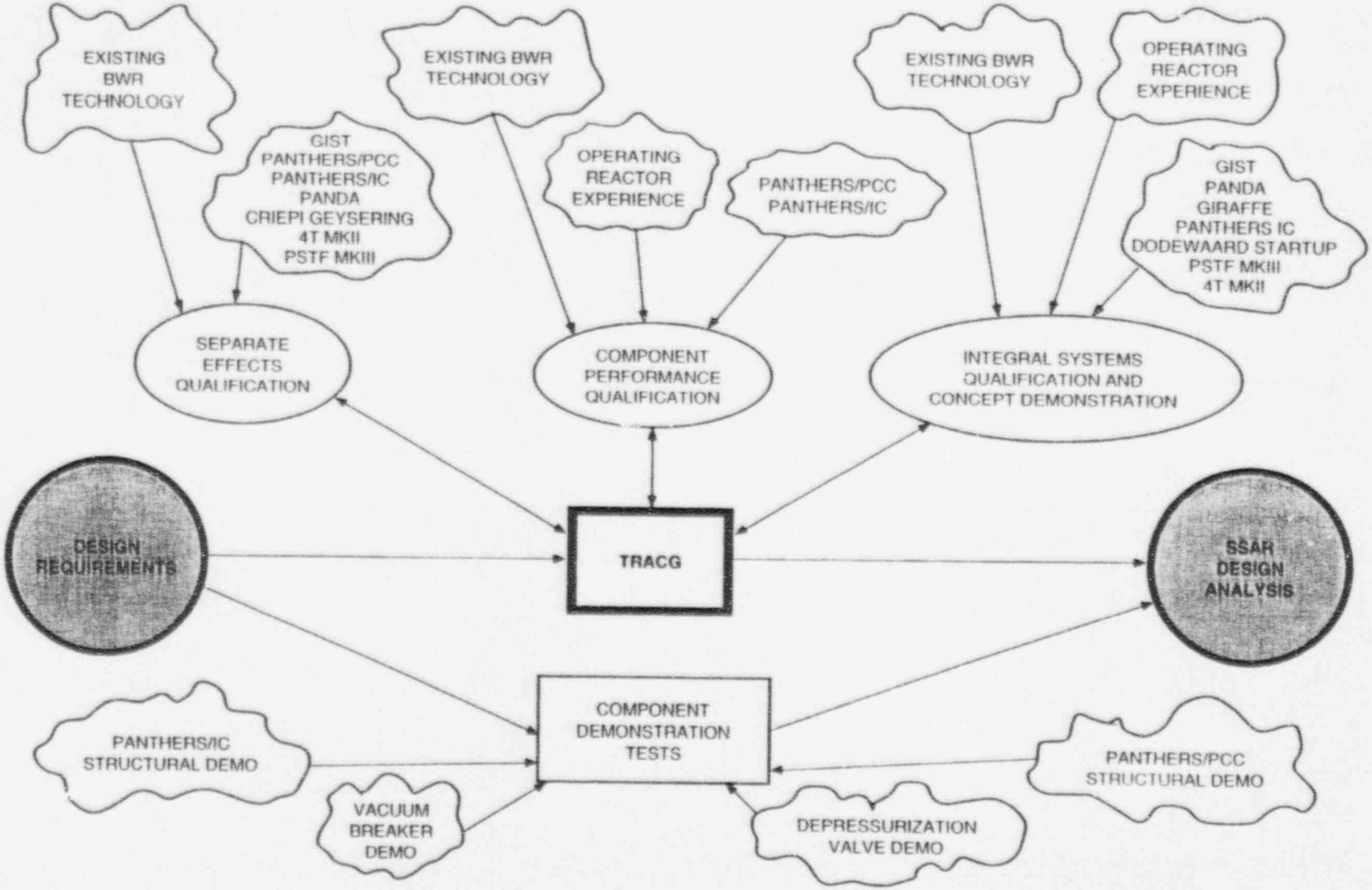


Figure 6.0-1. Technology Basis for SBWR Design

7.0 SUMMARY AND CONCLUSIONS

The Test and Analysis Program Description (TAPD) systematically defined tests 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 with 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). The scaling of the test facilities has been addressed quantitatively (Appendix B). This document supersedes previous 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 has been modified. The following summarizes the changes:

Test Plan

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

Analysis Plan

- GIST: Seven additional tests identified 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 tests from five other tests and one operating plant experience 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 Chapter 5 identifies elements which have been used to date, elements in which existing test data will be used, and elements in which forthcoming test data will be used to qualify the SBWR analytical model, TRACG. These are collected in Section 6 to show the composite TRACG qualification plan.

GE 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.

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Appendix A – Test and Analysis Plan (TAP)

A.1 Introduction

The study described in the main body of this document culminated in Table 6.1-1, which identified the qualification in addition to that presented in Reference [2] which is necessary for TRACG to be applied to SBWR safety analysis. This appendix identifies those specific tests and analyses that will be performed to meet the identified supplemental needs.

The overall 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 overall 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 test and analysis runs, definition of test types, and a 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

- Test matrices defining test conditions
- Test objectives
- Description of how the test data meets the specified objectives

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 197 steady-state test conditions, 14 transient performance demonstrations, and 38 integral systems tests. Section 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. Additionally, those test conditions chosen for analysis with TRACG are identified and

cross-referenced to the qualification needs. Section A.3.1.6 also gives an overview of other data that will be used for TRACG qualification beyond the qualification described in [2].

SBWR component performance tests are described in Section 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 over a range of accidents and transients, and then that range is bounded in the test matrix. Some SBWR tests are performed in a scaled configuration. In this case, 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 comprehensive Licensing Topical Report will be submitted documenting the testing, data analysis, and conclusions, following completion of all testing.

A.2.4 Analytical Approach

The analytical approach to be used is consistent with that previously documented in the TRACG Qualification Licensing Topical Report, NEDE-32177P [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. They will be documented and transmitted to the NRC prior to initiation of matrix testing. (This has already been done for the PANTHERS PCC thermal-hydraulic tests [35].)

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.3 Test and Analysis Plan

A.3.1 Thermal-Hydraulic Tests

A.3.1.1 PANTHERS/PCC

A.3.1.1.1 Test Description

PANTHERS/PCC (Passive Containment Cooling) testing is being performed as a joint study 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 will be performed in this facility. This section covers the thermal-hydraulic portion of the testing; component structural performance tests are covered in Section A.3.2.1.

The PCC condenser being tested 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 of the type being tested would be found in a SBWR. The PCC is installed in a water pool having the appropriate volume for one SBWR PCC assembly.

Figure A.3-2 is a schematic of the PANTHERS/PCC facility. Primary instrumentation is that required 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.

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 noncondensable 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

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 15 minutes. The time-averaged data are reported and analyzed.

Table A.3-1a 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) will be used to determine the baseline heat exchanger performance over a range of saturated steam flow rates without the presence of noncondensable gases. Test Group P1 data will be 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.3-1b shows the PANTHERS/PCC Steady-State Performance Matrix for Air-Steam Mixture Tests (150 individual test conditions are specified). As noted previously, the independent variables are steam mass flow rate, air mass flow rate, steam superheat conditions, and absolute operating pressure. Figure A.3-3 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 noncondensable 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 noncondensable 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, 24, 31, 32, 33, and 34 (Test Group P6) are lower priority tests. They may be run to supplement the previously identified tests by increasing the data density within the already established air/steam flow map.

Transient Test Conditions

PANTHERS/PCC transient tests will be used to establish noncondensable buildup effects and PCC pool water level effects. They are not intended to be systems transient tests.

Table A.3-1c shows the PANTHERS/PCC noncondensable buildup test matrix. Six test conditions are specified as Test Group P7. In these test conditions, steam will be supplied at a constant rate, and steady-state conditions established, similar to what was done in the steady-state performance tests. Air, helium, or air/helium mixtures will then be injected into the steam supply, with the vent line closed and the transient degradation in heat transfer performance will be measured, as a function of the total noncondensable mass injected.

- Tests Conditions 50 and 51 provide a baseline condition with air as the only noncondensable. Air is similar to nitrogen in molecular weight, and is heavier than steam. Test Conditions 75 and 76 are repeats of Test Conditions 50 and 51, but utilize helium as the noncondensable gas instead of air. Helium is lighter than steam, and will mix in a manner similar to hydrogen. The results of these four tests 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 effects of a combination of air and helium concurrently flowing into the heat exchanger.

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

Table A.3-1d shows the PANTHERS/PCC Pool Water Level Effect Test Matrix. Three test conditions are specified as Test Group P8. In these test conditions, steam and air/steam mixtures will be supplied to the PCC heat exchanger, and steady-state conditions established, similar to the steady-state performance tests. In these tests, however, the water level in the PCC pool will be allowed to drop and the PCC tubes to uncover. Both the PCC pool level and the PCC heat rejection rate will be monitored as a function of time.

- Test Conditions 54, 55, and 56 will establish the effects 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 may be compared to Test Conditions 41, 15, and 25, respectively, to obtain the effects of lowered water level on condensation performance. Test Conditions 54 and 55 may 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 TRACG Analysis Plan

Table A.3-2 lists those PANTHERS/PCC tests that will be analyzed with TRACG, cross-referenced to the qualification needs. 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.3-4 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.3-2. For the steady-state saturated and superheated steam conditions, the assessment of adequacy will be made on the *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 noncondensable 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 noncondensable 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.

Noncondensable Density – Analysis of Test Conditions 51 and 76 addresses the buildup of noncondensables 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

PANTHERS/IC (Isolation Condenser) testing will be performed at Societa Informazioni Esperienze Termoidrauliche (SIET) in Piacenza, Italy. The tests will be 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 will be performed during these tests. This section covers the thermal-hydraulic portion of the testing; component performance tests are covered in Section 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.3-5 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.

Figure A.3-6 is a schematic of the PANTHERS/IC facility. Primary instrumentation is that required to ascertain heat exchanger thermal-hydraulic performance, by performing mass and energy balances on the facility.

A.3.1.2.2 Test Objectives

The test 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 noncondensable venting capability of the SBWR IC design, and condensation restart capability following venting. *(Concept Demonstration)*

A.3.1.2.3 Test Matrix and Data Analysis

Steady-State Performance Tests

As for PANTHERS/PCC, the majority of the IC tests are steady-state performance tests. Table A.3-3a provides the PANTHERS/IC Steady-State Performance Test Matrix. Ten test conditions are specified. For these tests, the steam generator and IC module will be pressurized to an inlet pressure of 8.618 MPa, and the IC drain valves opened to initiate IC operation. When the

steam generator pressure has reached desired test condition inlet pressure, the facility will be stabilized at that pressure, and heat transfer performance data taken for a period of between fifteen minutes and one-half hour. The time averaged steady-state data will be reported and analyzed.

- 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 be used to demonstrate the startup of the IC heat exchanger under full-scale thermodynamic conditions. These tests are designed to demonstrate heat exchanger performance; they are not intended to be systems simulations.

Table A.3-3b gives the PANTHERS/IC Transient Demonstration Test Matrix. Four Test Conditions are specified. These tests will be performed in much the same manner as the steady-state performance tests, but transient data will be recorded over the course of the experiment. 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 air injected slowly after the steam generator pressure has been reduced to the value specified as "inlet pressure" in Table A.3-3b. The IC will be vented when the inlet pressure reaches 7.653 MPa (1110 psig) or when the pressure peaks, if at a lower value. Re-establishment of condensation following venting will be recorded. Test Condition 16 (Test Group I4) is a repeat of Test Condition 1, 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 the IC following venting of noncondensables. Test Condition 16 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 meet Test Objective 2.
- Test Group I2 demonstrates restart of the IC and meets Test Objective 3.
- Test Group I3 demonstrates restart of the IC following venting, and meets Test Objective 4.

A.3.1.2.4 TRACG Analysis Plan

Table A.3-4 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.3-4. 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 1.38 MPa) (1150 to 200 psig).

Noncondensable Buildup and Venting – Analysis of Test Conditions 12 and 13 demonstrates TRACG capability to predict the effect of noncondensable 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 Condition 16 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

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 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 scale, but reduced in number from the prototype. 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.

Figure A.3-7 shows a schematic of the PANDA test facility. 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 piping connections such that a PCC heat exchanger may be tested in a quasi-steady manner.

The PANDA data acquisition system is capable of recording up to 720 channels with each channel recorded once every two seconds. Sufficient instrumentation and data recording will be installed to characterize the thermal-hydraulic containment performance for tests lasting up to 24 hours.

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

A series of steady-state tests will be conducted using one of the PANDA PCC condensers. 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 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.3-5a shows the PANDA Steady-State PCC Performance test matrix. Six test conditions are included. 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.3-5a 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 constant air flows and a constant steam flow of 0.20 kg/sec. In addition, one test will be run with a pure steam flow equivalent to that expected to match the steam condensing capacity of the condenser (Test Condition S6).
- PANDA Test Conditions S1 through S6 provide a data base for TRACG qualification to meet the requirements of Test Objective 1(a).
- The results of PANDA Test Conditions S1 through S6 will be compared with the PANTHERS and GIRAFFE steady state performance data as noted in Table A.3-5a to meet the requirements of Test Objective 1(b).

Transient Integral Systems Tests

A series of 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. Table A.3-5b provides the test matrix summarizing the key characteristics of each test, and data use:

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

- *Test M1* is a simulation of a break in the main steamline of the SBWR. The initial conditions in the containment will be the same as those tested in the GIRAFFE Phase 2 main steamline break test. These initial conditions are similar to SBWR containment conditions consisting of a mixture of air and steam at one hour into the LOCA. One-third of the steam from the break will be directed to drywell DW1 which has one PCC condenser, and two-thirds of the steam will be directed to DW2 with two PCC condensers. These test conditions represent a symmetrical situation in the PANDA facility

  PANDA test M1 will be compared with GIRAFFE Phase 2 Main Steamline Break Test to assess the effects of facility scale on integral system performance.
- *Test M2* is a repeat of Test M1 with all of the break flow steam directed into drywell DW2. DW2 has two PCC condensers, and this test maximizes the steam content of DW2 and the air content of DW1. It is the most asymmetric condition that can be set in PANDA. Test M2 results will be compared with Test M1 results to quantify asymmetric effects on PCCS containment performance.
- *Test M3* is very similar to Test M1, but with nominal initial containment conditions as calculated for the SBWR under SSAR assumptions at one hour into the LOCA. The initial drywell pressure will be approximately 300 kPa (43.5 psi). This test will provide a base case for comparison to all other tests.
- *Test M4* is a repeat of test M3 to demonstrate transient system response repeatability.

- *Test M5* is a repeat of M3, but with continuous water supply flow to the RPV. In the SBWR, if AC power is available, an operator might use the Fuel and Auxiliary Pools Cooling System, for example, to provide active core cooling. This test will demonstrate potential systems interaction effects for cases with continuous cold water addition to the RPV. Continuous water supply flow to the RPV will result in filling of the RPV and flow of relatively cold water out of the break and into the drywell. This, in turn, is expected to result in opening of the drywell-to-wetwell vacuum breakers and reintroduction of air from the wetwell into the drywell. This test will provide data on PCC performance when air is reintroduced to the heat exchangers following essentially pure steam operation.
- *Test M6* is a repeat of Test M3 but 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 demonstrate the PCC condenser startup characteristics when initially blanketed with noncondensable gas.
- *Test M8* is a repeat of Test M3, but with drywell-to-wetwell bypass leakage. This test will provide the effect of bypass leakage on containment performance.
- *Test M9* is a combination of tests M5 and M7, with cold water injection into the RPV, and this test will begin as early as possible in the SBWR LOCA scenario, consistent with the PANDA facility design.
- PANDA tests M1 through M9 provide a database for TRACG qualification that meets Test Objective 2.
- PANDA tests M1 through M9 address long-term operation of the PCCS. Tests M5 through M7 and M9 address systems interaction and PCCS restart issues. These tests meet the requirements of Test Objective 3.

A.3.1.3.4 TRACG Analysis Plan

All six PANDA steady-state and all nine PANDA integral systems tests will have TRACG analysis performed. Analyses will be performed post-test or both pre- and post-test.

- ***Pure Steam Condensation*** – Analysis of Tests S1 and S6 demonstrates TRACG capability to predict pure saturated steam condensation rates at and above rated conditions.
- ***Air-Steam Mixtures*** – Analysis of Tests S2 through S5 addresses the effects of non-condensable mass fraction in the PANDA PCC configuration.
- ***Drywell-Wetwell Noncondensable Distribution*** – Analysis of tests M1, M2, and M5 through M9 addresses the effects of initial gas and vapor distribution within the containment system, including vacuum breaker flow, and demonstrate TRACG 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 through M9 address TRACG capability to model systems interactions.
- ***Bypass Leakage*** – TRACG analysis of test M8 provides qualification of the 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.3-8 provides a facility schematic, and Figure A.3-9 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.3-7. Twenty-six test conditions were specified. These 26 individual tests were divided into four test types, three of them loss-of-coolant accidents (LOCAs):

- Bottom Drain Line Break (BDLB)
- Main Steamline 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 case 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 the 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 case of a relatively small break below the core. Seven cases were run in this configuration.
- *Series MSLB* (Main Steamline Break) consisted of eight tests, six of which were parametric variations and two of which were repeat-performed to establish 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. These are among the most interesting runs, since they form a data set at or outside the limits of SBWR potential, 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. Perhaps the major difference between the GIST and final 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 meets Test Objective 1.
- The overall GIST database provides a sufficient basis for TRACG qualification and meets the requirements of Test Objective 2.

A3.1.4.4 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 main 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. Results are reported in Reference 2.

GIST tests which have had TRACG analysis completed are identified in Table A.3-8. These tests represent a full variety of break types, a wide range of initial liquid inventories in the pressure vessel, variations of containment initial conditions and a variety in the degree of availability of GDCS.

Table A.3-9 defines the additional cases that will be performed. In all cases, comparisons will performed on *GDCS flow rate*, *GDCS initiation time*, and *RPV pressure*.

A.3.1.5 GIRAFFE

A.3.1.5.1 Test Description

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 (IC/PCC) heat exchanger, and the connecting piping. Separate vessels represented the reactor pressure vessel (RPV), drywell (D/W), wetwell (WW), Gravity-Driven Cooling System pools (GDCS pools) and the IC/PCC pools, which house the IC/PCC condenser unit. A schematic of the facility is shown in Figure A.3-10.

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.3-11 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.

Measurements were taken throughout the system for absolute pressure, differential pressure, temperature, and flow rate. Instrument locations are given in Reference 38.

A.3.1.5.2 Test Objectives

The test 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 used are steady-state performance data of 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, and the condensed vapor and vented nitrogen were directed to volumes modeled to act as the reactor vessel (RPV) and suppression chamber pool (S/C), 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 10 minutes. The time averaged data were reported and analyzed.

Table A.3-10 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 G2 will be compared to that from corresponding PANDA and PANTHERS tests to corroborate those results at a third scale. Data from Test Group G1 provide a support database for TRACG qualification and meet the requirements of Test Objective 1.

System Response Tests

In the GIRAFFE system response tests, the RPV supplies steam to the drywell, simulating release of decay heat through a main steamline break. Steam and nitrogen flow from the drywell to the IC/PCC, with condensate returning to the RPV, and gases venting to the S/C.

Two GIRAFFE systems tests are useful in support of the test objectives. These two test conditions are identified in Table A.3-11, and comprise Data Group G2. The two tests identified

are a simulated Steamline Break Test from Phase 1 Step 3, and a similar steamline break test from Phase 2. The Phase 2 test is more like the current SBWR configuration, and has initial conditions which more closely resemble SBWR conditions under SSAR assumptions.

- Data from Test Group G2 will be compared to the corresponding PANDA test (GIRAFFE Phase 2 to PANDA M1) to address scale effects. Test Group G2 provides a support data base for TRACG PCCS performance and meets the requirements of Test Objective 2.

Recently, TOSHIBA has conducted a GIRAFFE integral system test with initial conditions similar to the Test Group G2 steamline break test from Phase 2, but with helium in place of nitrogen in the drywell. Initial conditions are specified in Table A.3-12. This test was not reported in [43]. It is identified as Test Group G3.

- Data from Test Group G3 provide a database for light density gas behavior. It may be used for TRACG PCCS performance prediction in this situation, and meets the requirement of Test Objective 2.

A3.1.5.4 TRACG Analysis Plan

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.

TRACG comparisons have been performed for all the Test Group G1 conditions, and for the Phase 1, Step 1 main steamline break test.

A.3.1.6 Other Analyses Planned

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

A.3.1.6.1 1/6 Scale Boron Mixing Test

GE Nuclear Energy has performed a set of boron mixing injection tests for the 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 on 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. These comparisons will address the mixing issues identified as qualification needs ATW1, ATW2, and ATW3.

A.3.1.6.2 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. These tests serve to address qualification needs C12 (natural circulation) and F4 (geysering during startup).

A.3.1.6.3 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, and 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. Comparisons with plant data will be made for *reactor pressure, downcomer subcooling, and downcomer pressure difference (a measure of recirculation flow)* to address qualification needs F4 and C12 of Table 6.1-1.

A.3.1.6.4 Containment System Response – PSTF Mark III

In the early 1970s, GE Nuclear Energy 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. Qualification needs XC7, MV1, MV3, and WW1 can be addressed by TRACG analysis of this data.

The test series chosen for these comparisons is PSTF Series 5703, which was reported in [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 steamline break size was the primary variable. Comparisons will be made on *drywell pressure and wetwell pressure* to address qualification needs XC7 and WW1, *main vent flow rate* to address qualification need MV1, and *top vent clearing time* to address qualification need MV3.

A.3.1.6.5 Containment System Response – Mark II 4T

In the mid-1970s, GE Nuclear Energy conducted a series of containment tests supporting the Mark II containment design in the 4T (Temporary Tall Test Tank) facility in San Jose, California. While the primary focus of this testing was suppression pool dynamics, three of the tests runs are particularly useful in addressing TRACG qualification needs XC7, WW2, and WW5.

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 135°F prior to test initiation to minimize steam condensation. One test, Run 35, used a unheated drywell. Very different response was seen due to steam condensation in the drywell. Additionally, two other runs, 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 runs will be analyzed with TRACG. *Drywell pressure* and *vent flow rate* from Run 33 will be compared to address qualification need WW5. *Drywell pressure, wetwell pressure, vent flow rate, and vacuum breaker opening/closing time* will be compared to address qualification need XC7.

A.3.1.6.6 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 5807 Run 29. Qualification need WW6 will be addressed by comparison of *suppression pool temperature distribution*.

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 being performed using the same hardware and test facility as described in Section A.3.1.1. The component demonstration tests will be 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.3-13. Data will be 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:

1. 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.3-14. 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 [18]. (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 will be performed during the PANTHERS/PCC component demonstration test: simulated LOCA pressurizations and simulated pneumatic leak test pressurizations.

Simulated LOCA Pressurizations

Simulated LOCA cycles will be performed by pressurizing the PCC units with steam, so that both the temperature and pressure effects of a LOCA are simulated. The PCC 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. Table A.3-15 gives the time history of the LOCA pressurizations. Each LOCA cycle will last approximately 30 minutes. Ten cycles will be performed.

Simulated Pneumatic Leak Test Pressurizations

Simulated pneumatic tests are performed by pressurizing the PCC heat exchanger with air to 758 kPa (110 psig). The PCC pool temperature will be at ambient conditions during these pressurizations. The test pressure will be held for 2 minutes for each cycle. A total of 300 cycles will be 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 PCC heat exchanger will be performed using the same hardware and test facility as described in Section 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 on Table A.3-16. 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:

1. Confirm that the mechanical design of the IC heat exchanger is adequate to assure its structural integrity over a 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 subject the equipment to a total number of pressure and temperature cycles well in excess of that expected over the anticipated duration of an SBWR ISI cycle. Specifically, it is planned to subject the IC to cycles equivalent to one-third of

the SBWR's 60 year lifetime (i.e., 20 years). Prototype non-destructive tests (NDT) will be performed before and after the cyclic testing. The test matrix is given as Table A.3-15. (Credit is taken for the thermal cycles experienced during the IC thermal-hydraulic testing in determination of this Component Demonstration Test Matrix.)

Simulated operational cycles will be performed by pressurizing the IC unit with steam, so that both the temperature and pressure effects of a LOCA are simulated. 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. 120 cycles will be performed. Data will be recorded for durations of several minutes, periodically through each cycle.

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. 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 in order 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., which is 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 in order to assess the effects of ambient temperature and steamline 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 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 will 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 are 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.
- Determine dynamic loads which result in lift of the disk.

- The opening and closing reliability are maintained after subjecting the fully aged valve to grit ingestion.

A3.2.4.3 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 flow stability and lift pressure met requirements. The flow test determined that the valve stroke was not sufficient to meet minimum flow requirements. Since the natural stability of the valve 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 would result 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 determined 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 leaktightness 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 is the reliability testing, which will subject the fully-aged valve to grit ingestion to simulate possible environmental conditions that could affect bearing surfaces and seals during normal service. The valve will be cycled three thousand times to demonstrate reliability at its required statistical failure rate of 10^{-4}.

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	18**	PCC steady-state performance; air/steam mixtures
PANTHERS/PCC	P7	6	PCC performance; noncondensable 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 startup demonstration
PANTHERS/IC	I3	2	IC restart demonstration, noncondensable venting
PANTHERS/IC	I4	1	IC performance; water level effects
PANDA/PCC	S	6	PCC steady-state performance; steam and air/steam mixtures
PANDA	Phase 1	2	Containment performance – basic
PANDA	Phase 2	7	Containment performance – integrated systems effects
GIRAFFE	G1	13	PCC steady-state performance – steam and air/steam mixtures
GIRAFFE	G2	2	Containment performance – integral system effects
GIRAFFE	G3	1	Containment performance – light density gas effects
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.3-1a. PANTHERS/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)	0 (0)
P1	38	1.4 (3.0)	0 (0)	0 (0)
P1	39	2.5 (5.5)	0 (0)	0 (0)
P1	40	3.6 (8.0)	0 (0)	0 (0)
P1	41	4.5 (10.0)	0 (0)	0 (0)
P1	42	5.7 (12.5)	0 (0)	0 (0)
P1	43	7.0 (15.4)	0 (0)	0 (0)
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	4.5 (10.0)	0 (0)	15 (27)
P2	48	4.5 (10.0)	0 (0)	20 (36)
P2	49	4.5 (10.0)	0 (0)	30 (54)

*Superheat conditions are relative to the steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P3	9-1	4.5 (10.0)	0.073 (0.16)	207 (30)	0 (0)
P3	9-2	4.5 (10.0)	0.073 (0.16)	328 (48)	0 (0)
P3	9-3	4.5 (10.0)	0.073 (0.16)	448 (65)	0 (0)
P3	9-4	4.5 (10.0)	0.073 (0.16)	569 (83)	0 (0)
P3	9-5	4.5 (10.0)	0.073 (0.16)	689 (100)	0 (0)
P3	15-1	4.5 (10.0)	0.14 (0.31)	193 (28)	0 (0)
P3	15-2	4.5 (10.0)	0.14 (0.31)	317 (46)	0 (0)
P3	15-3	4.5 (10.0)	0.14 (0.31)	441 (64)	0 (0)
P3	15-4	4.5 (10.0)	0.14 (0.31)	565 (82)	0 (0)
P3	15-5	4.5 (10.0)	0.14 (0.31)	689 (100)	0 (0)
P3	18-1	4.5 (10.0)	0.36 (0.79)	186 (27)	0 (0)
P3	18-2	4.5 (10.0)	0.36 (0.79)	274 (40)	0 (0)
P3	18-3	4.5 (10.0)	0.36 (0.79)	362 (53)	0 (0)
P3	18-4	4.5 (10.0)	0.36 (0.79)	450 (65)	0 (0)
P3	18-5	4.5 (10.0)	0.36 (0.79)	538 (78)	0 (0)
P3	23-1	4.5 (10.0)	0.83 (1.83)	159 (23)	0 (0)
P3	23-2	4.5 (10.0)	0.83 (1.83)	240 (35)	0 (0)
P3	23-3	4.5 (10.0)	0.83 (1.83)	321 (47)	0 (0)
P3	23-4	4.5 (10.0)	0.83 (1.83)	402 (59)	0 (0)
P3	23-5	4.5 (10.0)	0.83 (1.83)	483 (70)	0 (0)
P4	2-1	1.4 (3.0)	0.014 (0.030)	207 (30)	0 (0)
P4	2-2	1.4 (3.0)	0.014 (0.030)	328 (48)	0 (0)
P4	2-3	1.4 (3.0)	0.014 (0.030)	448 (65)	0 (0)
P4	2-4	1.4 (3.0)	0.014 (0.030)	569 (83)	0 (0)
P4	2-5	1.4 (3.0)	0.014 (0.030)	689 (100)	0 (0)
P4	13-1	2.5 (5.5)	0.14 (0.31)	193 (28)	0 (0)
P4	13-2	2.5 (5.5)	0.14 (0.31)	283 (41)	0 (0)
P4	13-3	2.5 (5.5)	0.14 (0.31)	373 (54)	0 (0)
P4	13-4	2.5 (5.5)	0.14 (0.31)	462 (67)	0 (0)

*Superheat referenced to steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P4	13-5	2.5 (5.5)	0.14 (0.31)	552 (80)	0 (0)
P4	16-1	7.0 (15.4)	0.14 (0.31)	200 (29)	0 (0)
P4	16-2	7.0 (15.4)	0.14 (0.31)	322 (47)	0 (0)
P4	16-3	7.0 (15.4)	0.14 (0.31)	445 (65)	0 (0)
P4	16-4	7.0 (15.4)	0.14 (0.31)	567 (82)	0 (0)
P4	16-5	7.0 (15.4)	0.14 (0.31)	689 (100)	0 (0)
P4	17-1	2.5 (5.5)	0.36 (0.79)	172 (25)	0 (0)
P4	17-2	2.5 (5.5)	0.36 (0.79)	255 (37)	0 (0)
P4	17-3	2.5 (5.5)	0.36 (0.79)	338 (49)	0 (0)
P4	17-4	2.5 (5.5)	0.36 (0.79)	420 (61)	0 (0)
P4	17-5	2.5 (5.5)	0.36 (0.79)	503 (73)	0 (0)
P4	19-1	5.7 (12.5)	0.36 (0.79)	193 (28)	0 (0)
P4	19-2	5.7 (12.5)	0.36 (0.79)	283 (41)	0 (0)
P4	19-3	5.7 (12.5)	0.36 (0.79)	373 (54)	0 (0)
P4	19-4	5.7 (12.5)	0.36 (0.79)	462 (67)	0 (0)
P4	19-5	5.7 (12.5)	0.36 (0.79)	552 (80)	0 (0)
P4	22-1	1.4 (3.0)	0.83 (1.83)	97 (14)	0 (0)
P4	22-2	1.4 (3.0)	0.83 (1.83)	161 (23)	0 (0)
P4	22-3	1.4 (3.0)	0.83 (1.83)	225 (33)	0 (0)
P4	22-4	1.4 (3.0)	0.83 (1.83)	288 (42)	0 (0)
P4	22-5	1.4 (3.0)	0.83 (1.83)	352 (51)	0 (0)
P4	25-1	7.0 (15.4)	0.83 (1.83)	179 (26)	0 (0)
P4	25-2	7.0 (15.4)	0.83 (1.83)	262 (38)	0 (0)
P4	25-3	7.0 (15.4)	0.83 (1.83)	345 (50)	0 (0)
P4	25-4	7.0 (15.4)	0.83 (1.83)	427 (62)	0 (0)
P4	25-5	7.0 (15.4)	0.83 (1.83)	510 (74)	0 (0)
P5	35-1	4.5 (10.0)	0.83 (1.83)	159 (23)	20(36)
P5	35-2	4.5 (10.0)	0.83 (1.83)	240 (35)	20(36)
P5	35-3	4.5 (10.0)	0.83 (1.83)	321 (47)	20(36)

*Superheat referenced to steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P5	35-4	4.5 (10.0)	0.83 (1.83)	402 (58)	20(36)
P5	35-5	4.5 (10.0)	0.83 (1.83)	483 (70)	20(36)
P5	36-1	4.5 (10.0)	0.83 (1.83)	159 (23)	30(54)
P5	36-2	4.5 (10.0)	0.83 (1.83)	240 (35)	30(54)
P5	36-3	4.5 (10.0)	0.83 (1.83)	321 (47)	30(54)
P5	36-4	4.5 (10.0)	0.83 (1.83)	402 (58)	30(54)
P5	36-5	4.5 (10.0)	0.83 (1.83)	483 (70)	30(54)
P6	1-1	0.45 (1.0)	0.014 (0.030)	193 (28)	0 (0)
P6	1-2	0.45 (1.0)	0.014 (0.030)	317 (46)	0 (0)
P6	1-3	0.45 (1.0)	0.014 (0.030)	441 (64)	0 (0)
P6	1-4	0.45 (1.0)	0.014 (0.030)	565 (82)	0 (0)
P6	1-5	0.45 (1.0)	0.014 (0.030)	689 (100)	0 (0)
P6	3-1	2.5 (5.5)	0.027 (0.060)	207 (30)	0 (0)
P6	3-2	2.5 (5.5)	0.027 (0.060)	328 (48)	0 (0)
P6	3-3	2.5 (5.5)	0.027 (0.060)	448 (65)	0 (0)
P6	3-4	2.5 (5.5)	0.027 (0.060)	569 (83)	0 (0)
P6	3-5	2.5 (5.5)	0.027 (0.060)	689 (100)	0 (0)
P6	4-1	3.6 (8.0)	0.027 (0.060)	207 (30)	0 (0)
P6	4-2	3.6 (8.0)	0.027 (0.060)	328 (48)	0 (0)
P6	4-3	3.6 (8.0)	0.027 (0.060)	448 (65)	0 (0)
P6	4-4	3.6 (8.0)	0.027 (0.060)	569 (83)	0 (0)
P6	4-5	3.6 (8.0)	0.027 (0.060)	689 (100)	0 (0)
P6	5-1	4.5 (10.0)	0.027 (0.060)	207 (30)	0 (0)
P6	5-2	4.5 (10.0)	0.027 (0.060)	328 (48)	0 (0)
P6	5-3	4.5 (10.0)	0.027 (0.060)	448 (65)	0 (0)
P6	5-4	4.5 (10.0)	0.027 (0.060)	569 (83)	0 (0)
P6	5-5	4.5 (10.0)	0.027 (0.060)	689 (100)	0 (0)
P6	6-1	5.7 (12.5)	0.027 (0.060)	207 (30)	0 (0)
P6	6-2	5.7 (12.5)	0.027 (0.060)	328 (48)	0 (0)

*Superheat referenced to steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P6	6-3	5.7 (12.5)	0.027 (0.060)	448 (65)	0 (0)
P6	6-4	5.7 (12.5)	0.027 (0.060)	569 (83)	0 (0)
P6	6-5	5.7 (12.5)	0.027 (0.060)	689 (100)	0 (0)
P6	7-1	7.0 (15.4)	0.027 (0.060)	207 (30)	0 (0)
P6	7-2	7.0 (15.4)	0.027 (0.060)	328 (48)	0 (0)
P6	7-3	7.0 (15.4)	0.027 (0.060)	448 (65)	0 (0)
P6	7-4	7.0 (15.4)	0.027 (0.060)	569 (83)	0 (0)
P6	7-5	7.0 (15.4)	0.027 (0.060)	689 (100)	0 (0)
P6	8-1	1.4 (3.0)	0.073 (0.16)	193 (28)	0 (0)
P6	8-2	1.4 (3.0)	0.073 (0.16)	317 (46)	0 (0)
P6	8-3	1.4 (3.0)	0.073 (0.16)	441 (64)	0 (0)
P6	8-4	1.4 (3.0)	0.073 (0.16)	565 (82)	0 (0)
P6	8-5	1.4 (3.0)	0.073 (0.16)	689 (100)	0 (0)
P6	10-1	5.7 (12.5)	0.073 (0.16)	207 (30)	0 (0)
P6	10-2	5.7 (12.5)	0.073 (0.16)	328 (48)	0 (0)
P6	10-3	5.7 (12.5)	0.073 (0.16)	448 (65)	0 (0)
P6	10-4	5.7 (12.5)	0.073 (0.16)	569 (83)	0 (0)
P6	10-5	5.7 (12.5)	0.073 (0.16)	689 (100)	0 (0)
P6	11-1	7.0 (15.4)	0.073 (0.16)	207 (30)	0 (0)
P6	11-2	7.0 (15.4)	0.073 (0.16)	328 (48)	0 (0)
P6	11-3	7.0 (15.4)	0.073 (0.16)	448 (65)	0 (0)
P6	11-4	7.0 (15.4)	0.073 (0.16)	569 (83)	0 (0)
P6	11-5	7.0 (15.4)	0.073 (0.16)	689 (100)	0 (0)
P6	12-1	0.45 (1.0)	0.14 (0.31)	138 (20)	0 (0)
P6	12-2	0.45 (1.0)	0.14 (0.31)	212 (31)	0 (0)
P6	12-3	0.45 (1.0)	0.14 (0.31)	286 (42)	0 (0)
P6	12-4	0.45 (1.0)	0.14 (0.31)	360 (52)	0 (0)
P6	12-5	0.45 (1.0)	0.14 (0.31)	434 (63)	0 (0)
P6	14-1	3.6 (8.0)	0.14 (0.31)	193 (28)	0 (0)

*Superheat referenced to steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P6	14-2	3.6 (8.0)	0.14 (0.31)	317 (46)	0 (0)
P6	14-3	3.6 (8.0)	0.14 (0.31)	441 (64)	0 (0)
P6	14-4	3.6 (8.0)	0.14 (0.31)	565 (82)	0 (0)
P6	14-5	3.6 (8.0)	0.14 (0.31)	689 (100)	0 (0)
P6	20-1	4.5 (10.0)	0.59 (1.29)	179 (26)	0 (0)
P6	20-2	4.5 (10.0)	0.59 (1.29)	262 (38)	0 (0)
P6	20-3	4.5 (10.0)	0.59 (1.29)	345 (50)	0 (0)
P6	20-4	4.5 (10.0)	0.59 (1.29)	428 (62)	0 (0)
P6	20-5	4.5 (10.0)	0.59 (1.29)	510 (74)	0 (0)
P6	21-1	7.0 (15.4)	0.59 (1.29)	186 (27)	0 (0)
P6	21-2	7.0 (15.4)	0.59 (1.29)	274 (40)	0 (0)
P6	21-3	7.0 (15.4)	0.59 (1.29)	362 (53)	0 (0)
P6	21-4	7.0 (15.4)	0.59 (1.29)	450 (66)	0 (0)
P6	21-5	7.0 (15.4)	0.59 (1.29)	538 (78)	0 (0)
P6	24-1	5.7 (12.5)	0.83 (1.83)	165 (24)	0 (0)
P6	24-2	5.7 (12.5)	0.83 (1.83)	248 (36)	0 (0)
P6	24-3	5.7 (12.5)	0.83 (1.83)	331 (48)	0 (0)
P6	24-4	5.7 (12.5)	0.83 (1.83)	413 (60)	0 (0)
P6	24-5	5.7 (12.5)	0.83 (1.83)	496 (72)	0 (0)
P6	31-1	2.5 (5.5)	0.027 (0.060)	207 (30)	20(36)
P6	31-2	2.5 (5.5)	0.027 (0.060)	328 (48)	20(36)
P6	31-3	2.5 (5.5)	0.027 (0.060)	448 (65)	20(36)
P6	31-4	2.5 (5.5)	0.027 (0.060)	569 (83)	20(36)
P6	31-5	2.5 (5.5)	0.027 (0.060)	689 (100)	20(36)
P6	32-1	2.5 (5.5)	0.027 (0.060)	207 (30)	30(54)
P6	32-2	2.5 (5.5)	0.027 (0.060)	328 (48)	30(54)
P6	32-3	2.5 (5.5)	0.027 (0.060)	448 (65)	30(54)
P6	32-4	2.5 (5.5)	0.027 (0.060)	569 (83)	30(54)
P6	32-5	2.5 (5.5)	0.027 (0.060)	689 (100)	30(54)

*Superheat referenced to steam partial pressure.

Table A.3-1b. 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 g (psig)]	Superheat* [°C (°F)]
P6	33-1	7.0 (15.4)	0.027 (0.060)	207 (30)	20(36)
P6	33-2	7.0 (15.4)	0.027 (0.060)	328 (48)	20(36)
P6	33-3	7.0 (15.4)	0.027 (0.060)	448 (65)	20(36)
P6	33-4	7.0 (15.4)	0.027 (0.060)	569 (83)	20(36)
P6	33-5	7.0 (15.4)	0.027 (0.060)	689 (100)	20(36)
P6	34-1	7.0 (15.4)	0.027 (0.060)	207 (30)	30(54)
P6	34-2	7.0 (15.4)	0.027 (0.060)	328 (48)	30(54)
P6	34-3	7.0 (15.4)	0.027 (0.060)	448 (65)	30(54)
P6	34-4	7.0 (15.4)	0.027 (0.060)	569 (83)	30(54)
P6	34-5	7.0 (15.4)	0.027 (0.060)	689 (100)	30(54)

*Superheat referenced to steam partial pressure.

Table A.3-1c. PANTHERS/PCC Noncondensable – Buildup Test Matrix

Test Group Number	Test Condition Number	Steam Flow [kg/s (lb/s)]	Helium Flow [kg/s (lb/s)]	Air Flow [kg/s (lb/s)]	Superheat* [°C (°F)]
P7	50	1.4 (3.0)	0 (0)	low	0 (0)
P7	51	4.5 (10.0)	0 (0)	low	0 (0)
P7	75	1.4 (3.0)	low	0 (0)	0 (0)
P7	76	4.5 (10.0)	low	0 (0)	0 (0)
P7	77	1.4 (3.0)	low	3.4 x He	0 (0)
P7	75	4.5 (10.0)	low	3.4 x He	0 (0)

*Superheat referenced to steam partial pressure.

Table A.3-1d. 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	4.5 (10.0)	0 (0)	0 (0)
P8	55	4.5 (10.0)	0.14 (0.31)	0 (0)
P8	56	7.0 (15.4)	0.83	0 (0)

*Superheat referenced to steam partial pressure.

Table A.3-2. 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.3-3a. PANTHERS/IC Steady-State Performance – Test Matrix

Test Condition Number	No. of Cycles	Test Group No.	Inlet Pressure [MPa g (psig)]	Initial Pool Temp. [°C (°F)]
2	1	I1	7.920 (1150)	<32 (90)
3	1	I1	7.240 (1050)	<32 (90)
4	1	I1	6.21 (900)	<32 (90)
5	1	I1	5.52 (800)	<32 (90)
6	1	I1	4.83 (700)	<32 (90)
7	1	I1	4.14 (600)	<32 (90)
8	1	I1	3.45 (500)	<32 (90)
9	1	I1	2.76 (400)	<32 (90)
10	1	I1	2.07 (300)	<32 (90)
11	1	I1	1.38 (200)	<32 (90)

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

Test Condition Number	No. of Cycles	Test Group Number	Initial Pressure, P1 MPag (psig)	Inlet Pressure [MPa g (psig)]	Initial Pool Temp. [°C (°F)]
1	2	I2	9.480 (1375)	8.618 (1250)	<21 (70)
12	1	I3	8.618 (1250)	0.48 (70)	<32 (90)
13	1	I3	8.618 (1250)	2.08 (300)	<32 (90)
16	1	I4	8.618 (1250)	8.618 (1250)	<32 (90)

Table A.3-4. 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
16	Pre/Post	Heat Rejection Rate

Table A.3-5a. PANDA Steady-State PCC Performance Test Matrix

PANDA Test No.	Steam Flow (kg/s)	Air Flow (kg/s)	PANTHERS Test Condition No.	GIRAFFE Phase 1, Step 1 Test No.
S1	0.20	0	41	2
S2	0.20	0.003	9	4
S3	0.20	0.006	15	6
S4	0.20	0.016	18	8
S5	0.20	0.03	23	10
S6	0.26	0	43	3

Table A.3-5b. PANDA Integral Systems Test Matrix

Panda Test No.	Break Type	No. of PCC	RPV Water Supply Flow	No. of IC	Bypass Leakage Area	Initial Conditions	Comments
M1	MSL -33% to DW1 -67% to DW2	1 in DW1 2 in DW2	0	0	0	GIRAFFE	Repeat of Giraffe Phase 2 MSLB Test
M2	MSL -0% to DW1 -100% to DW2	1 in DW1 2 in DW2	0	0	0	GIRAFFE	Repeat of Giraffe Phase 2 MSLB Test with asymmetric steam flow to DW1 and 2
M3	Same as M1	1 in DW1 2 in DW2	0	0	0	SSAR	Repeat of M1 with SSAR conditions
M4	Same as M1	1 in DW1 2 in DW2	0	0	0	SSAR	Repeat of M3
M5	Same as M1	1 in DW1 2 in DW2	Yes	0	0	SSAR	Repeat of M3 with continuous RPV injection water
M6	Same as M1	1 in DW1 2 in DW2	0	1	0	SSAR	Repeat of M3 with IC
M7	Same as M1	1 in DW1 2 in DW2	0	0	0	PCC filled with air, early start	Repeat of M3 with PCC blanketed with air
M8	Same as M1	1 in DW1 2 in DW2	0	0	TBD	SSAR	Repeat of M3 with DW to WW bypass leakage
M9	Same as M1	1 in DW1 2 in DW2	Yes	0	0	SSAR, early start	Cold water injection to open vacuum breaker

Table A.3-6. 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
M1	Pre/Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
		PCC Flows
M2	Pre/Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
M3	Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
M4	Same as M3	

Table A.3-6. PANDA TRACG Analysis Cases (Continued)

Test Number	Pre/Post Test	Data Comparison
M5	Pre/Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
		PCC Flows
		Vacuum Breaker Flow
M6	Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
		PCC Flows
		IC Flow
M7	Pre/Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
		PCC Flows
M8	Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp

Table A.3-6. PANDA TRACG Analysis Cases (Continued)

Test Number	Pre/Post Test	Data Comparison
M9	Pre/Post	Drywell Pressure
		Wetwell Pressure
		Drywell Temp
		Wetwell Temp
		Suppression Pool Temp
		PCC Flows

Table A.3-7. 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 Flow	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 RPV 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	59.2	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
C04 High LP DPV Setpt.	2	347	373	88	5	11.5	67.2	105	7.0

Table A.3-7. 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)
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	67.2	157	0.0
D06 Low GDCS Injection	4	347	865	74	0	0	67.2	107	0.0
D07 No Power	3	347	—	0	0	0	67.2	107	0.0

[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 SBWR. Used to determine decay heat.

Table A.3-8. 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.3-9. Additional GIST Runs for TRACG Analysis

Run	Type
B03	MSLB, Low Suppression Pool Level
A01	BDLB, Base Case
A03	BDLB, Max GDCS Flow Area
A05	BDLB, CRD Flow
D01A	NB, Base Case
D02	NB, Max GDCS Flow Area
D04	NB, Pressurized Wetwell

Table A-3.10. 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	200
13	G145	0.03	0.02	400

Table A-3.11. GIRAFFE System Response Tests (Test Group G2)

	Parameter	S/C	RPV	IC/PCC Pool	D/W
PHASE 1 STEP 3	**Pressure Total (MPa)**	0.301	0.314		0.314
	N2 Partial Pressure (MPa)	0.278			0.016
	Temperature (deg C)	63	134	100	133
	Level (m)	5.55	11	3.55	0
PHASE 2 MSL Break	**Pressure Total (MPa)**	0.174	0.193		0.192
	N2 Partial Pressure (MPa)	0.164			0.054
	Temperature (deg C)	53	119	100	108
	Level (m)	5.8	8.1*	3.55	0.0

*Elevation above GDCS nozzle

Table A.3-12. GIRAFFE Helium Test Conditions (Test Group G3)

Parameter	S/C	RPV	D/W
Total Pressure (MPa)	0.174	0.189	0.188
He Partial Press			0.135
N2 Partial Press			0.053
Temperature (deg C)	53	118	108

Table A.3-13. PCC Containment Cooler Structural Instrumentation

Measurement/Location	No. of Positions	Quantity at each Position	Total Meas.	Direct
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:				
inlet elbow	1	2	2	axial
inlet/header junction	1	2	2	Z
upper header/tube junction	5	1 or 2	7	Z
tube/lower header junction	3	1	3	Z
lower header	2	2	4	X, Y
lower header cover	1	2	2	Z, X
upper header	2	4	8	X, Z
upper header cover	1	4	4	X, Z
upper header cover bolts	3	1 or 2	5	Y
lower header cover bolts	3	1 or 2	5	Y
drain/lower header junction	1	2	2	X, Z
lower header supports	1	2	2	Z
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.3-14. 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
Pneu. Test	300	758	Ambient	2

Table A.3-15. LOCA Cycle Time History

PCC Inlet Pressure kPa (psig)	Time to Reach Pressure Sec
0 (0)	0
175 (25.4)	<2.3
249 (36.1)	<32
261 (37.8)	<67
379 (55)	<30 minutes

Table A.3-16. Isolation Condenser Structural Measurements

Measurement/Location	No. of Positions	Quantity at each Position	Total Meas.	Direct
Acceleration:				
mid-length of tube	5	2	10	X, Y
drain line curve	1	3	3	X, Y, Z
lower header cover	1	1	1	Z
upper header cover	1	5	3	X, Y, Z
Displacement:				
steam distributor	1	1	1	Z
drain/lower header junction	1	1	1	Z
steam pipe lower zone	1	1	21	Z
Total Strain:				
inlet/upper header junction	1	6	6	X, Y, Z
upper header/tube junction	5	1 or 2	7	Z
mid-length of tube	3	1	3	circ.
tube/lower header junction	3	1	3	Z
lower header	2	2	4	X, Y
lower header cover	1	2	2	X, Z
upper header	2	4	8	X, Y
upper header cover	1	4	4	X, Z
drain/lower header junction	1	4	4	X, Z
drain line curve	1	2	2	Y
drain line/drain tube	1	4	4	X
upper header cover bolts	3	2 or 1	5	Y
lower header cover bolts	3	2 or 1	5	Y
guard pipe/distributor	1	3	3	X, Z
support	1	2	2	X, 45°
upper header near support	1	4	4	X, Y
Permanent strain:				
inlet/header junction	1	3	3	Y, Z, 45°
upper header/tube junction	3	1	3	Z
lower header/drain junction	1	1	2	Z
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.4-17. IC Component Demonstration Test Matrix

Test Cond. No.	No. of Cycles	Cycle Type	Initial Pressure MPa g (psig)	Inlet Pressure MPa g (psig)	Initial Pool Temp. °C (°F)
1	1	1	9.480 (1375)	8.618 (1250)	<21 (70)
2	1	2	8.618 (1250)	7.920 (1150)	<32 (90)
3	1	2	8.618 (1250)	7.240 (1050)	<32 (90)
4	1	2	8.618 (1250)	6.21 (900)	<32 (90)
5	1	2	8.618 (1250)	5.52 (800)	<32 (90)
6	1	2	8.618 (1250)	4.83 (700)	<32 (90)
7	1	2	8.618 (1250)	4.14 (500)	<32 (90)
8	1	2	8.618 (1250)	3.45 (500)	<32 (90)
9	1	2	8.618 (1250)	2.76 (400)	<32 (90)
10	1	2	8.618 (1250)	2.07 (300)	<32 (90)
11	1	2	8.618 (1250)	1.38 (200)	<32 (90)
12	2	3	8.618 (1250)	0.48 (70)	<32 (90)
13	2	3	8.618 (1250)	2.07 (300)	<32 (90)
14	3	3	8.618 (1250)	4.83 (700)	<32 (90)
15	1	3	8.618 (1250)	7.24 (1050)	<32 (90)
16	15	1	8.618 (1250)		<32 (90)
17	85	4	8.618 (1250)		<32 (90)
18	1	5	9.480 (1375)	8.618 (1250)	<32 (90)

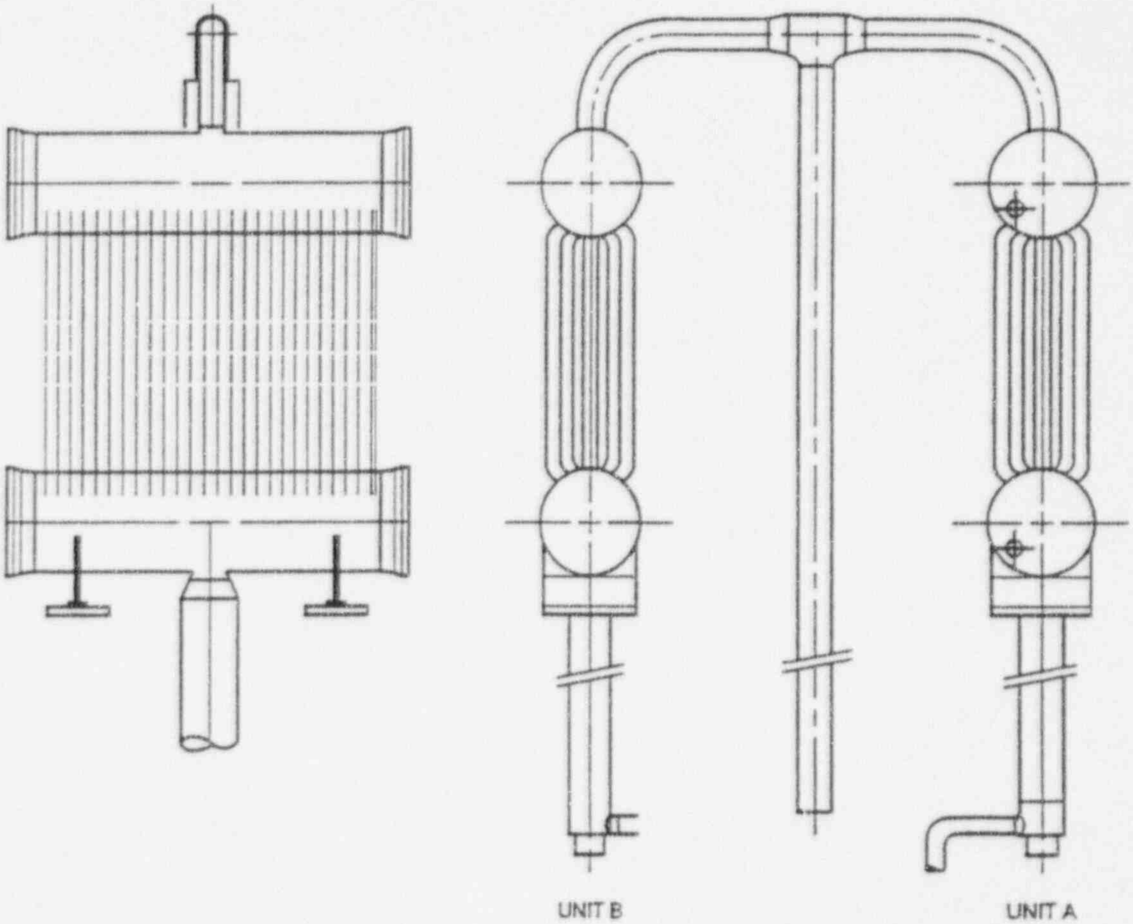

Figure A.3-1. Passive Containment Cooler Test Article

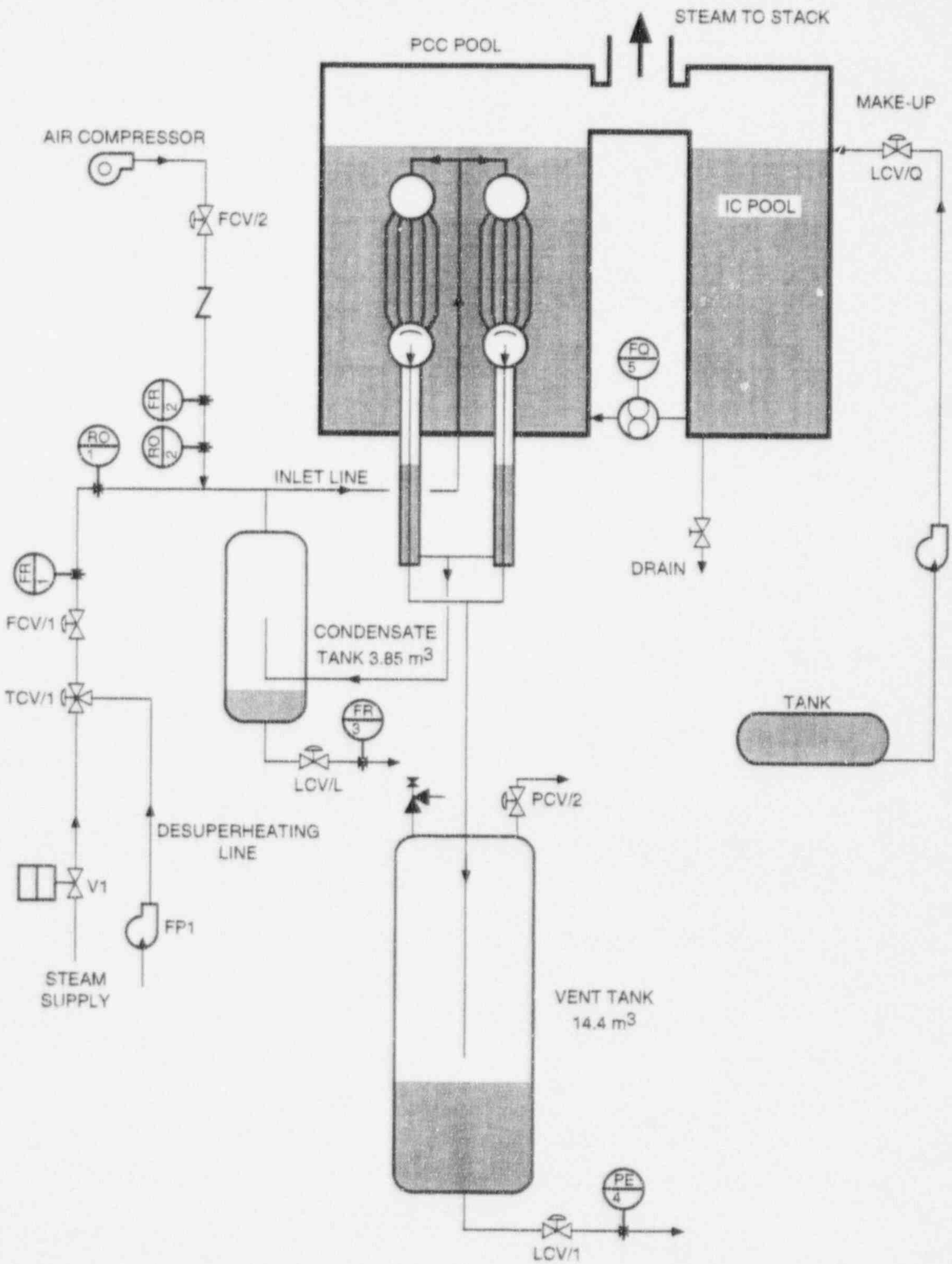

Figure A.3-2. PANTHERS/PCC Test Facility Schematic

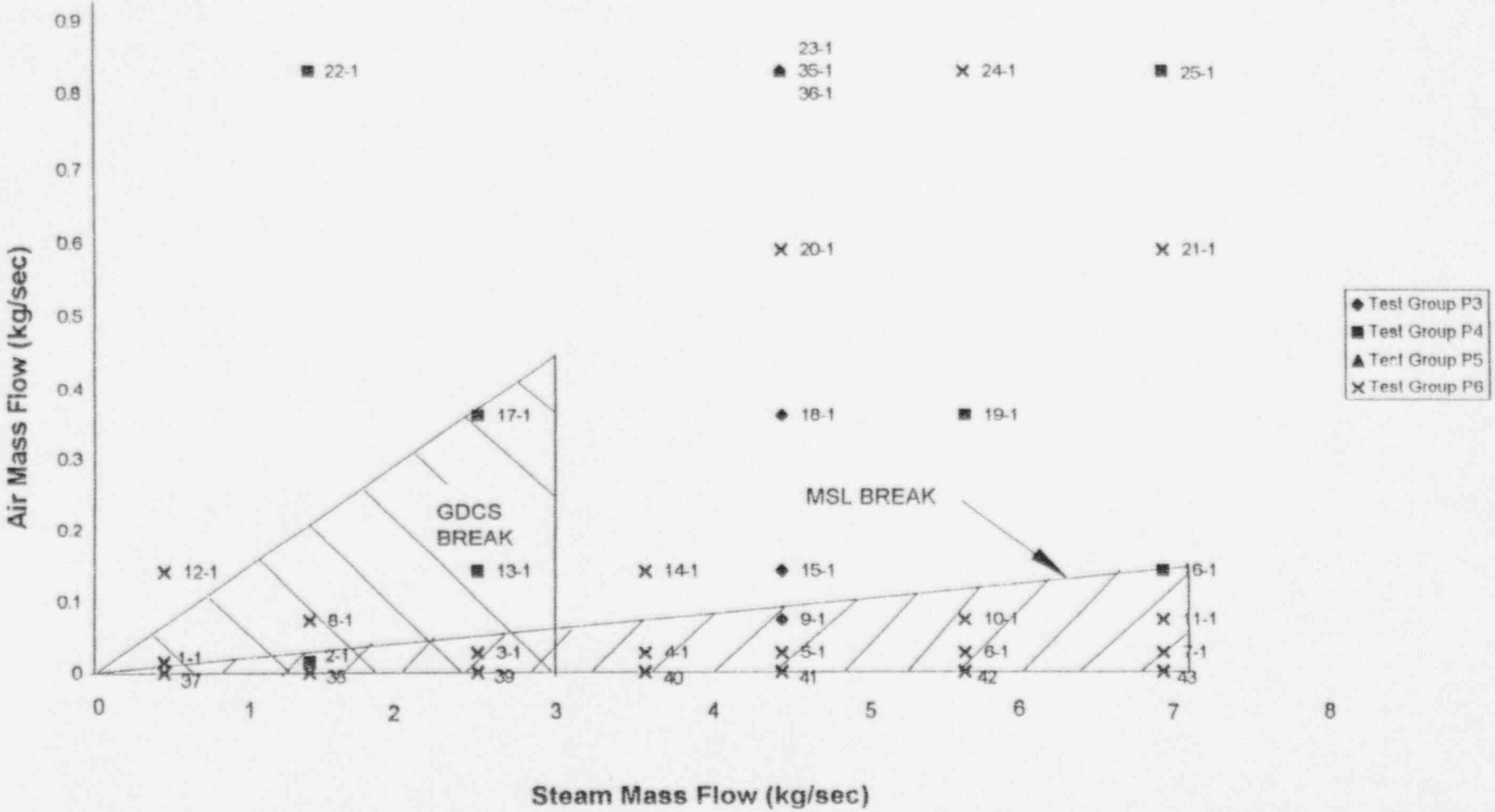

Figure A.3-3. Comparison of PANTHERS/PCC Steam-Air Test Range to SBWR Condition

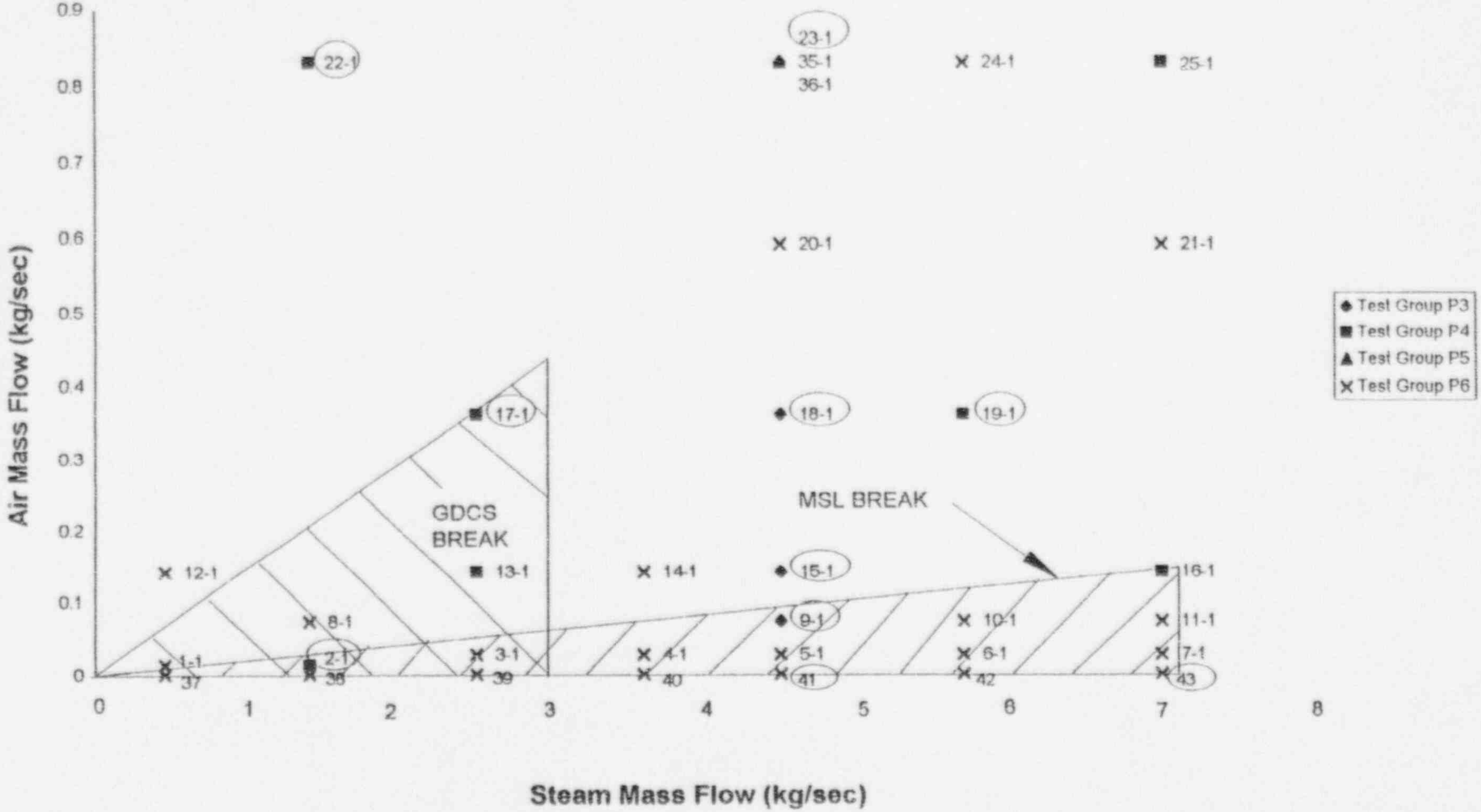

Figure A.3-4. TRACG PANTHERS/PCC Qualification Points

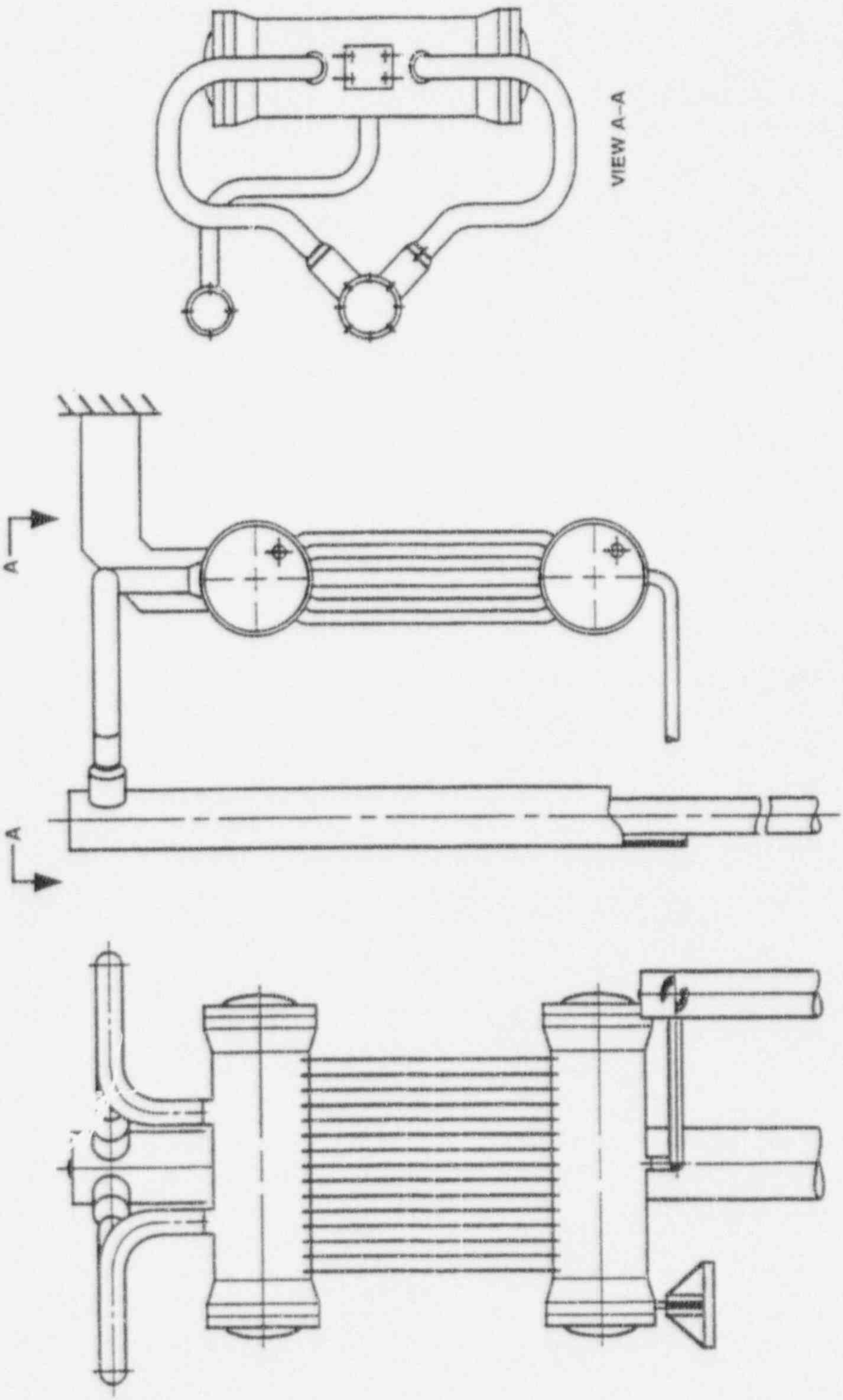

Figure A.3-5. Isolation Condenser Test Article

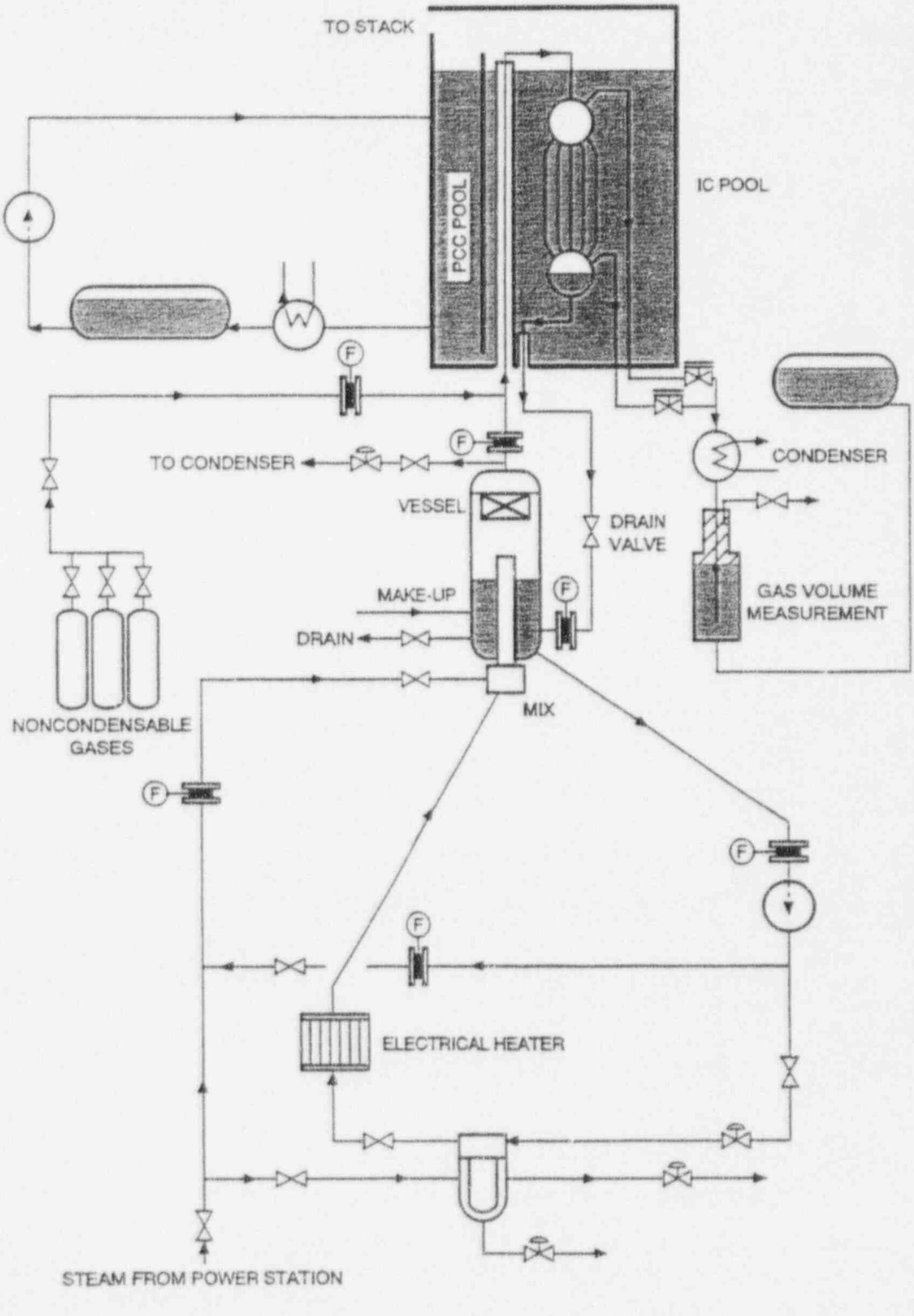

Figure A.3-6. PANTHERS/IC Test Facility Process Diagram

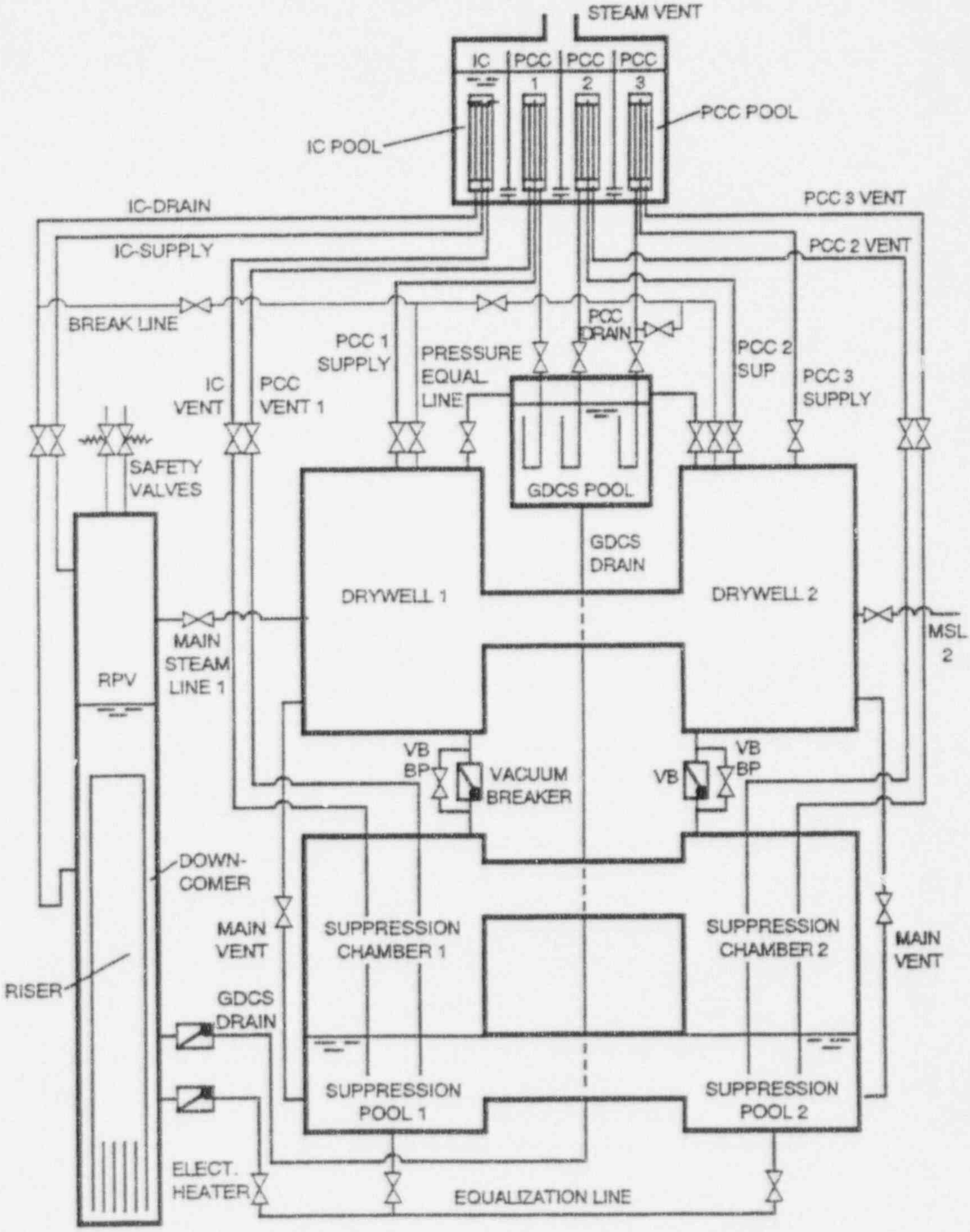

Figure A.3-7. PANDA Facility Schematic

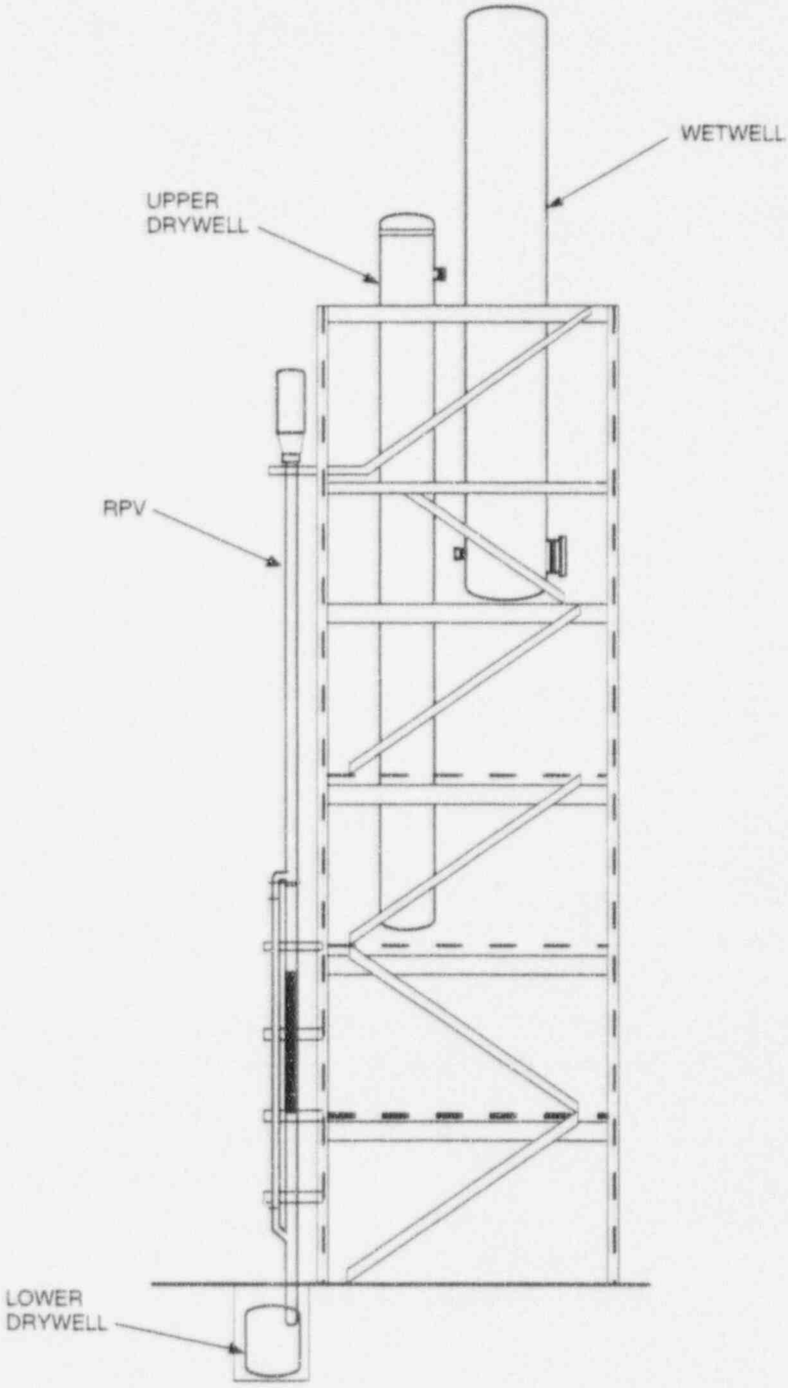

Figure A.3-8. GIST Facility Schematic

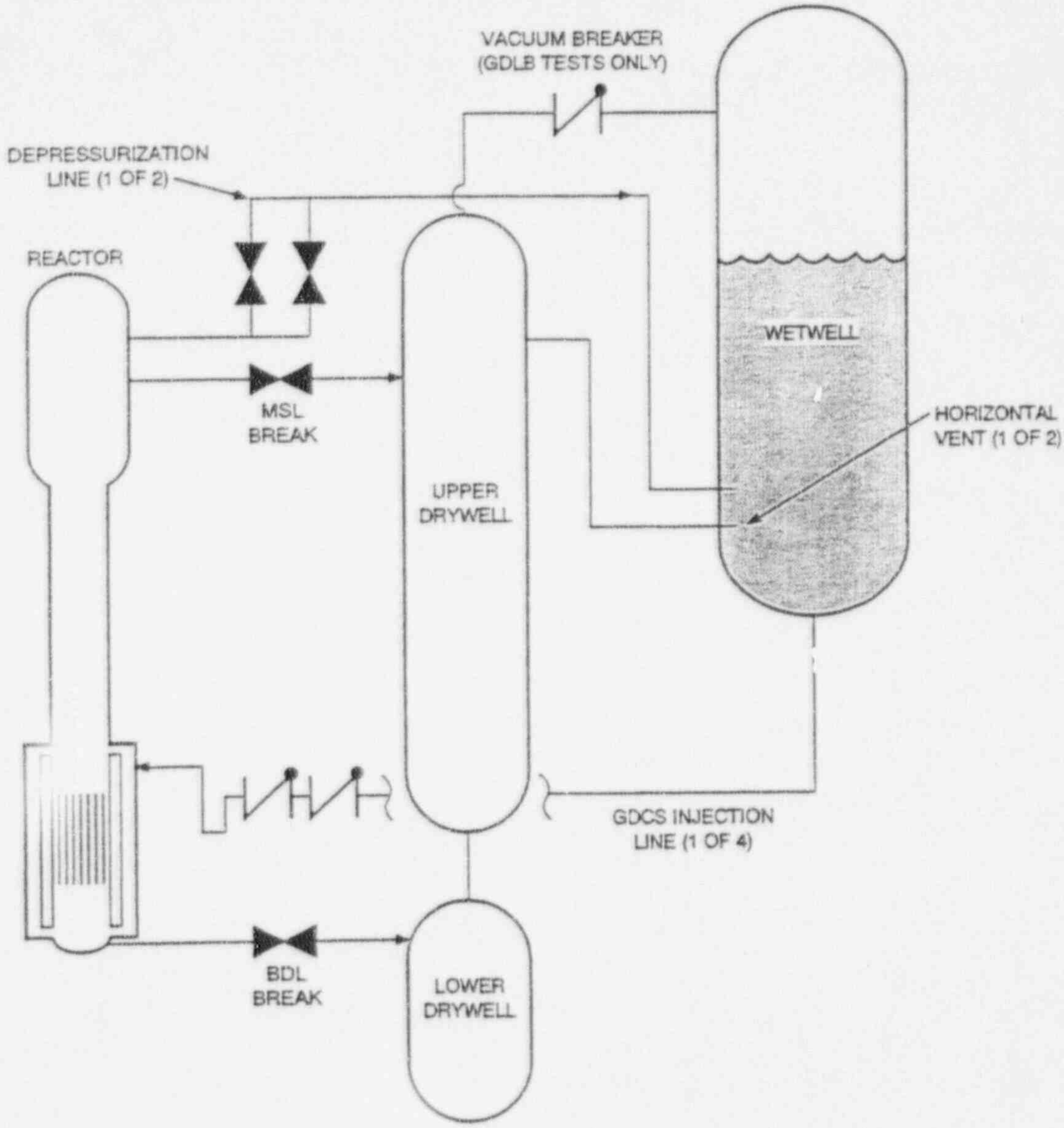

Figure A.3-9. GIST Facility Piping Arrangement

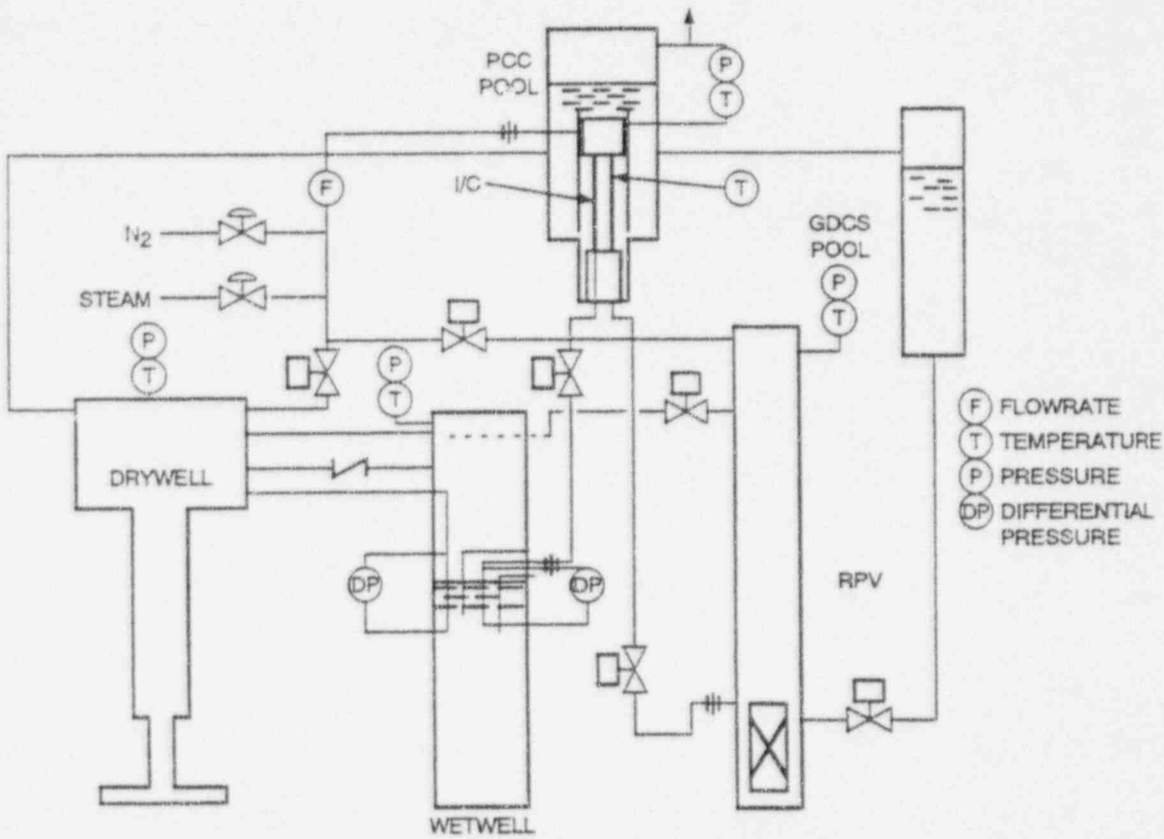

Figure A.3-10. GIRAFFE Test Facility Schematic

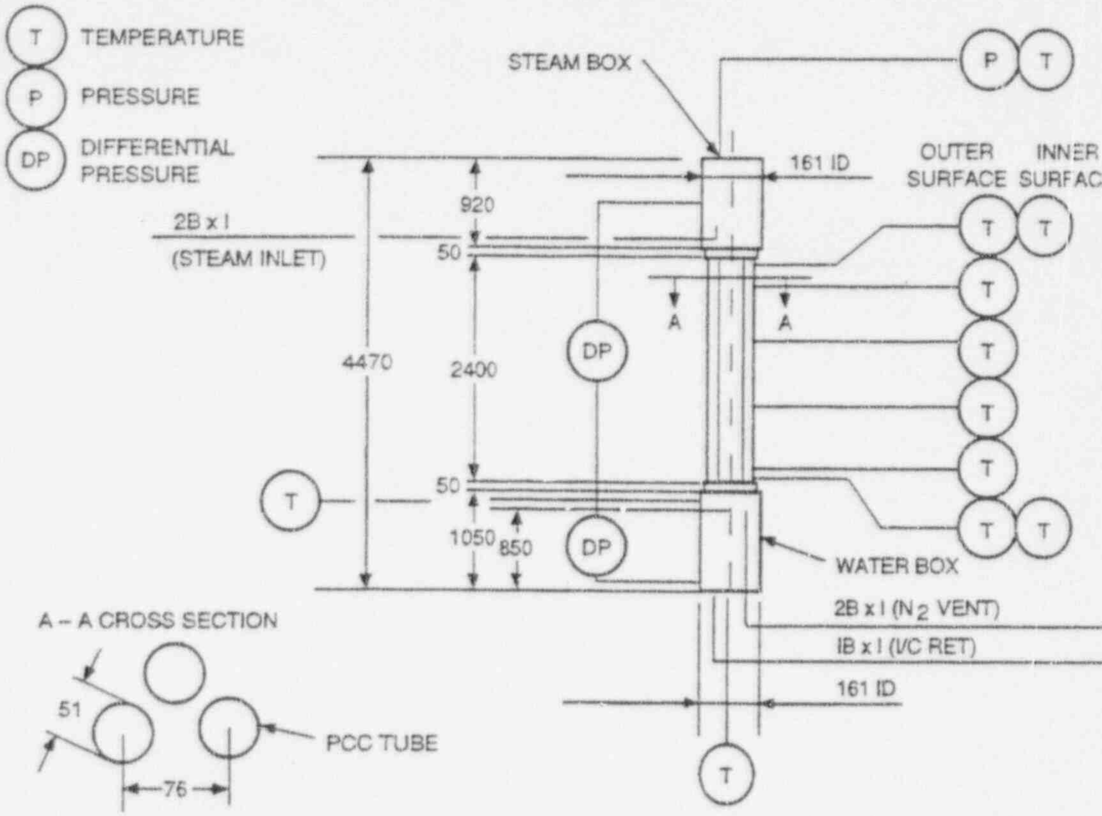

Figure A.3-11. GIRAFFE PCC Unit

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.

The scaling reported in Reference 32 follows the Hierarchical, Two-Tiered Scaling (H2TS) 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."

B.2 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).

B.3 Scaling of GIST Facility

B.3.1 Facility Description and Test Characteristics

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

The system scale is 1:508 and the facility is composed of the following regions:

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

The ICS and PCCS 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. 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 final SBWR 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) used in the SBWR. A complete discussion of the differences is contained in the appendix of Reference 42.

B.4 Scaling of GIRAFFE Facility

B.4.1 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.

B.5 Scaling of PANDA Facility

B.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.

B.6 PANTHERS Scaling

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, Section A.3.1.2.

B.7 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.

- **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.

- **PANTHERS** – The PANTHERS tests are full-scale component tests of the PCC and IC. 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.

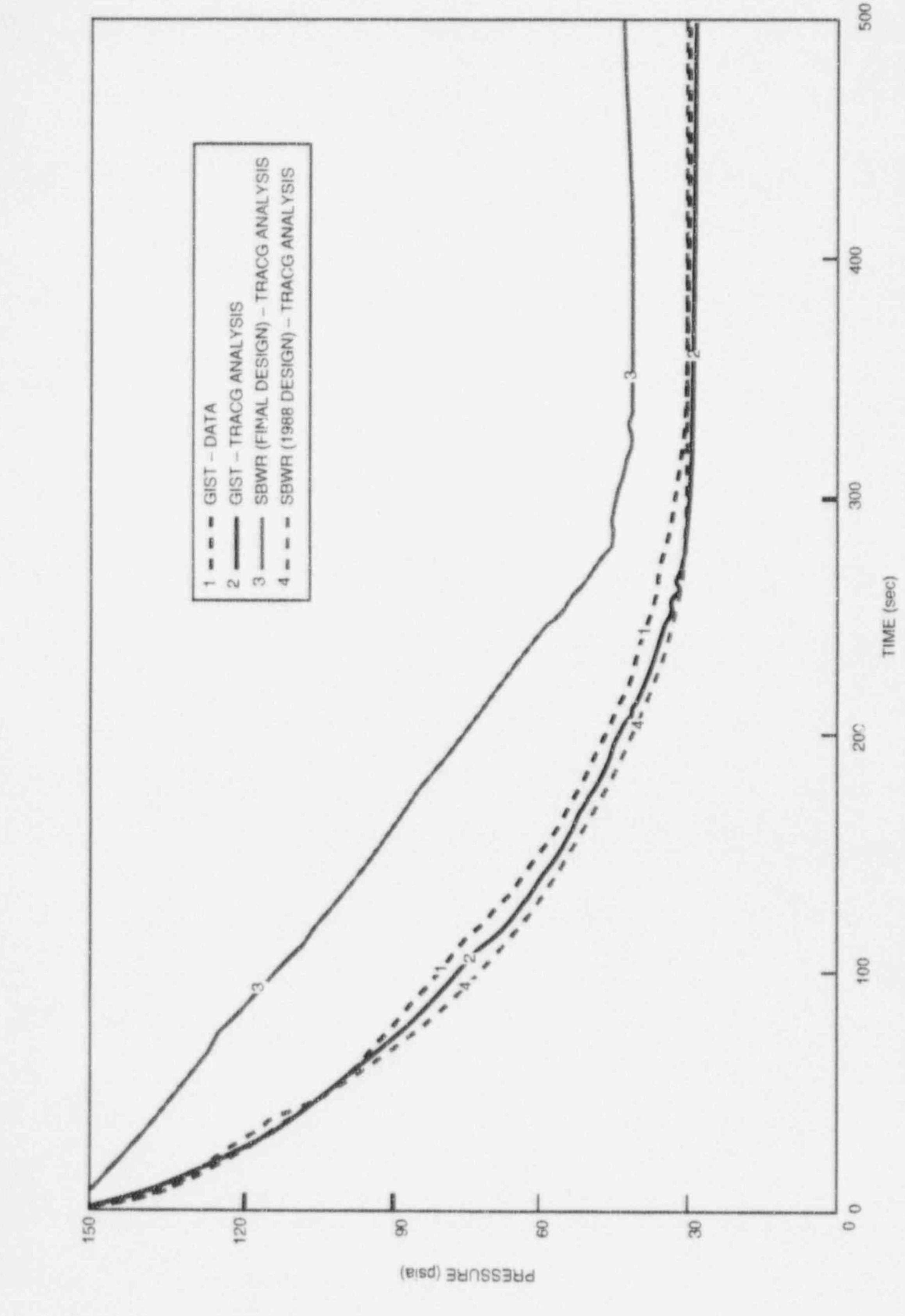

Figure B.3-1. Comparison of RPV Pressure Response for MSLB in GIST and SBWR

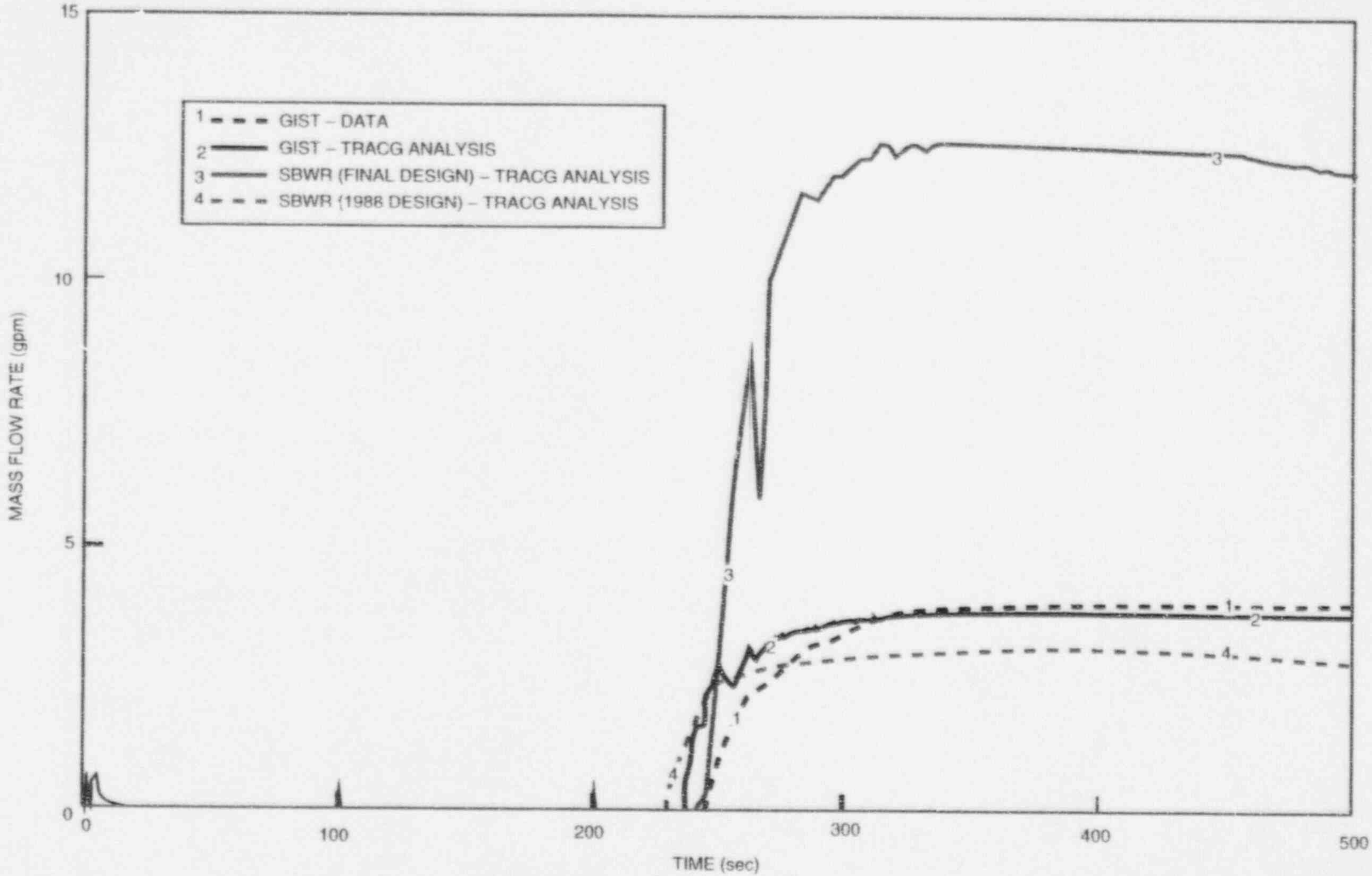

Figure B.3-2. Comparisons of GDCS Flow for MSLB in GIST and SBWR

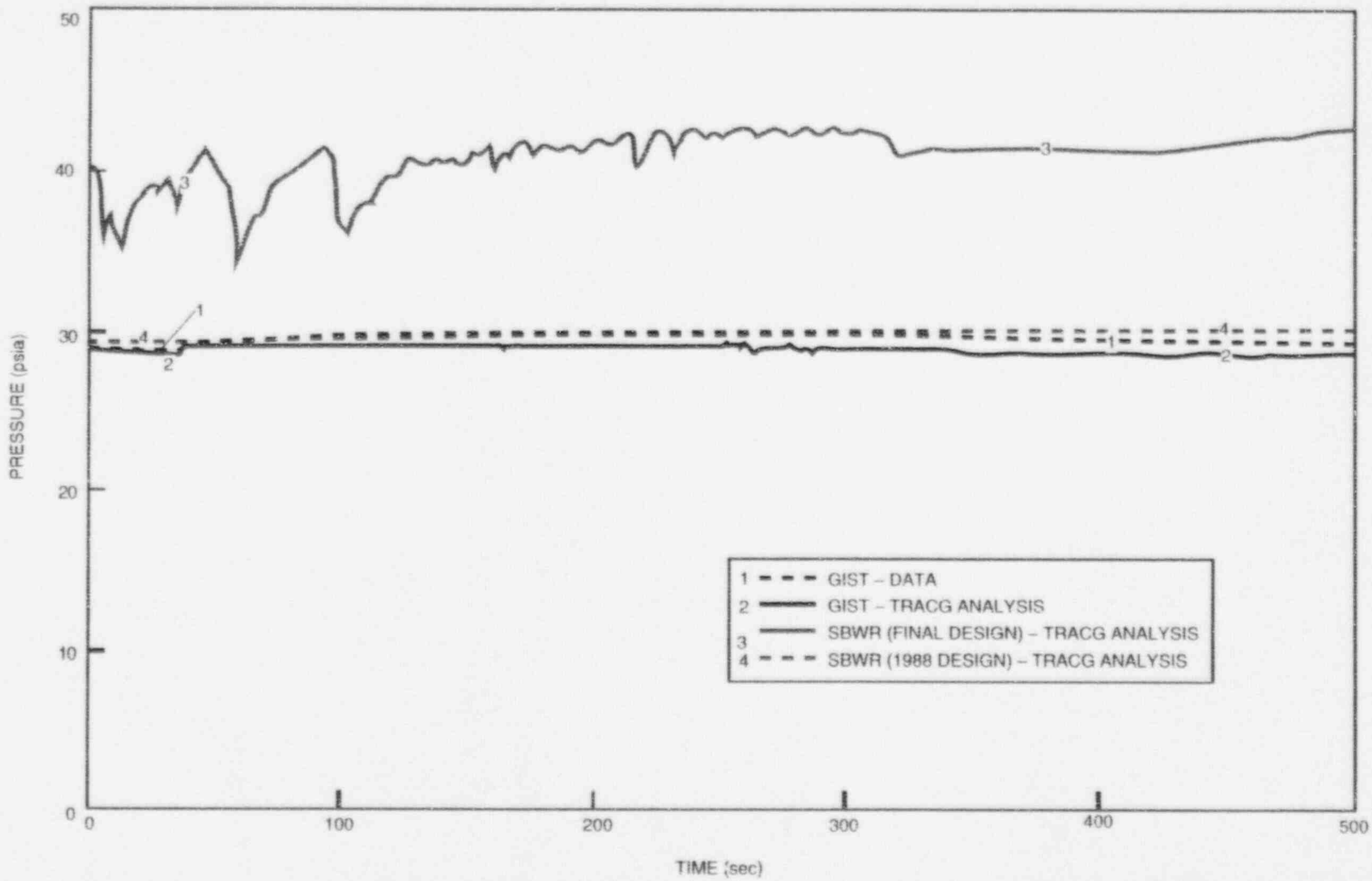

Figure B.3-3. Comparison of Drywell Pressure Response for MSLB in GIST and SBWR

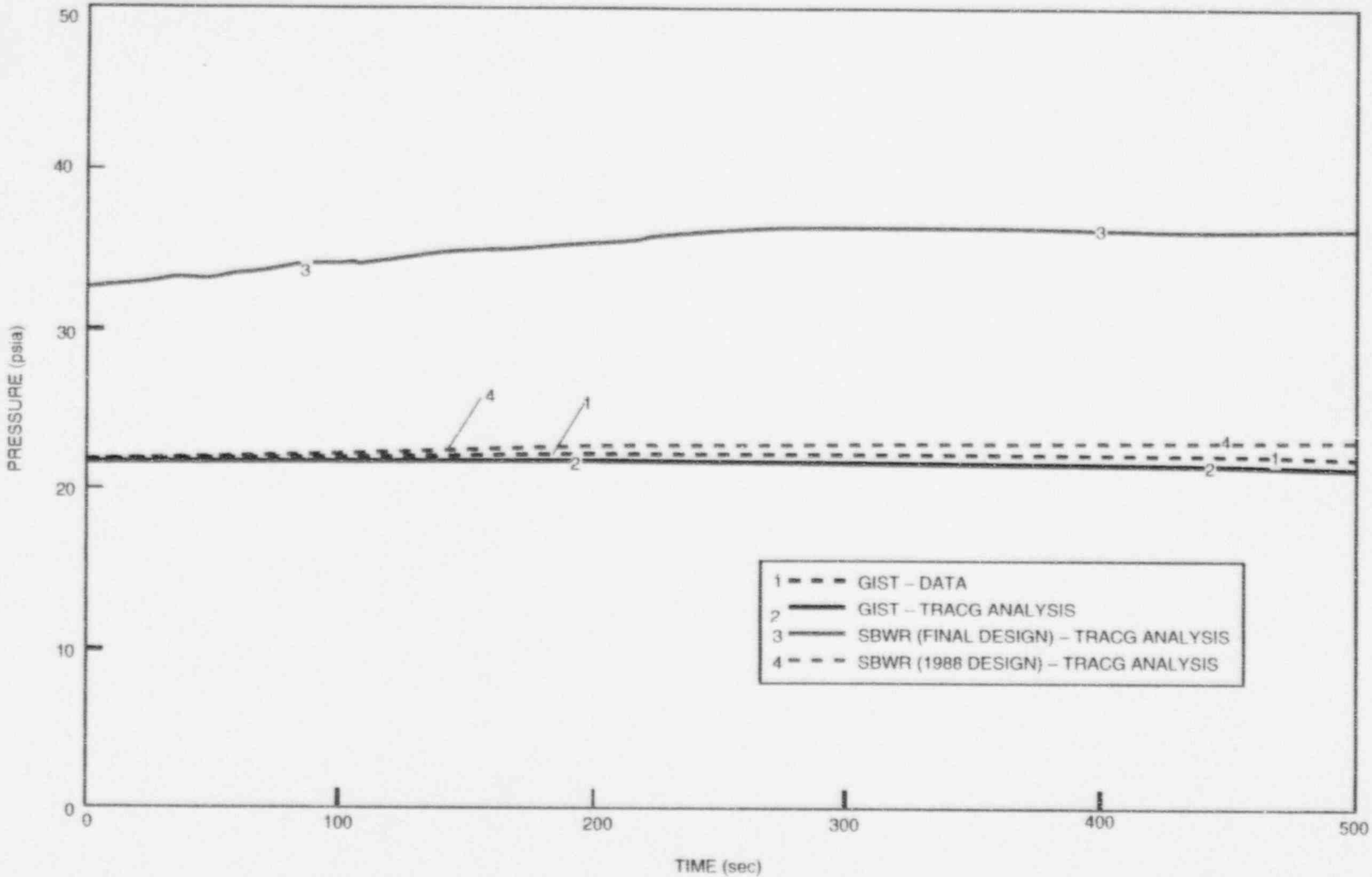

Figure B.3-4. Comparison of Wetwell Pressure Response for MSLB in GIST and SBWR

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	Volumetric flow rate [m^3/s]
l	Length [m]
L	Sum of lengths [m]
M	Mass [kg]
p	Pressure [Pa]
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/m3]
τ	Time constant [s]

Subscripts

G	Gas
L	Liquid

LG	Change liquid to gas
R	Scaling factor between prototype and model

Additional subscripts are defined in the text

Superscripts

o	Reference scale or 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 used in scaling the facilities.

B1-2 Top-Down Scaling

B1-2.1 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

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

Phase-change

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

Interfacial Phase Change

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

Inertial Pressure Drop

$$\Pi_{in} = \left\{ \frac{\rho^{o} {u_{r}^{o}}^{2}}{\Delta p^{o}} \right\}$$

Submergence

$$\Pi_{sub} = \left\{ \frac{\rho_{l} g H_{sub}^{o}}{\Delta p^{o}} \right\}$$

Hydrostatic Pressure Drop

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

Additionally, there are three time scales:

Volume time constant

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

Transit time constant

$$\tau_{tr} = \left\{ \frac{L_v}{u_r^{o}} \right\}; \quad L_V = \sum \frac{a_n}{a_r} l_n$$

Inertial time constant

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

and two geometric parameters:

Ratio of equivalent inertia and volume lengths

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

Total flow resistance

$$F = \sum F_n \frac{{a_r}^2}{{a_n}^2} + 2 \left\{ \frac{{a_r}^2}{{a_2}^2} - \frac{{a_r}^2}{{a_1}^2} \right\} ; \qquad F_n = \frac{4 f_n l_n}{D_n} + k_n$$

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 timescale.

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

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

Vaporization number

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

Pressure change time constant ratio

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

Phase Change Number

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

Depressurization Number

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

Flashing Number

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

Density ratio

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

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.

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 return 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, and 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 operated 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 different break locations were considered.

C.4 Containment Interaction Studies

The containment system interactions study investigated interactions between the ESF systems, and interactions of these 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 in this study included 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.