



GE Nuclear Energy

NEDO-32391
Revision C
DRF A70-00002
Class 1
August 1995

Licensing Topical Report

SBWR Test and Analysis Program Description



9509010267 950831
PDR ADOCK 05200004
A PDR



GE Nuclear Energy

J. E. Quinn, Projects Manager
LMR and SBWR Programs

General Electric Company
175 Curtner Avenue, M/C 165 San Jose, CA 95125-1014
408 925-1005 (phone) 408 925-3991 (facsimile)

August 31, 1995

MFN 169-95
Docket STN 52-004

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

Attention: Theodore E. Quay, Director
Standardization Project Directorate

Subject: **SBWR - Test and Analysis Program Description,
NEDO-32391, Revision C (Non-proprietary)**

Reference: 1. MFN 166-95, J. E. Quinn (GE) to T. E. Quay (NRC), *SBWR Test and Analysis Program Description*, NEDC-32391P, Revision C (Proprietary), August 28, 1995.

This letter transmits the non-proprietary version of Revision C of the *SBWR Test and Analysis Program Description (TAPD)* report, NEDO-32391P, for your review. This report provides a comprehensive, integrated plan that addresses the testing and analysis elements needed for analysis of SBWR performance.

Revision C replaces Revision B in total. The changes from Revision B to Revision C are identified by sidebars in the left-hand margins of the affected pages. Please dispose of all copies of Revision B or return them to GE Nuclear Energy, Attention: J. E. Leatherman, M/C 781.

Sincerely,

James E. Quinn,
Projects Manager

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

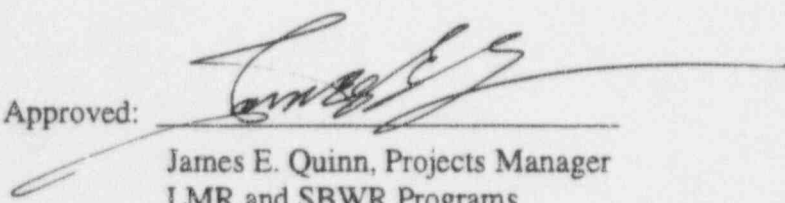
cc: P. A. Boehnert (NRC/ACRS) - [7 paper copies w/encl., plus E-Mail w/o encl.]
I. Catton (ACRS) - [1 paper copy w/encl., plus E-Mail w/o encl.]
S. Q. Ninh (NRC) - [21 paper copies w/encl., plus E-Mail w/o encl.]
J. H. Wilson (NRC) - [1 paper copy w/encl., plus E-Mail w/o encl.]

NEDO-32391
Revision C
DRF A70-00002
Class 1
August, 1995

SBWR Test and Analysis Program Description

B. S. Shiralkar
T. R. McIntyre
J. M. Healzer
W. Marquino
H. A. Upton
R. E. Gamble
J. R. Fitch
Y. C. Chu
P.F. Billig
J.C. Shaug

Approved: _____


James E. Quinn, Projects Manager
LMR and SBWR Programs

**IMPORTANT NOTICE REGARDING
CONTENTS OF THIS REPORT
PLEASE READ CAREFULLY**

The only undertakings of the General Electric Company (GE) respecting information in this document are contained in the contract between the customer and GE, as identified in the purchase order for this report and nothing contained in this document shall be construed as changing the contract. The use of this information by anyone other than the customer or for any purpose other than that for which it is intended, is not authorized; and with respect to any unauthorized use, GE makes no representation or warranty, and assumes no liability as to the completeness, accuracy, or usefulness of the information contained in this document.

IDENTIFICATION OF CHANGES

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

TABLE OF CONTENTS

1.0	INTRODUCTION.....	1-1
1.1	Purpose.....	1-1
1.1.1	Scope.....	1-1
1.2	Background.....	1-4
1.2.1	Use of TRACG.....	1-5
1.2.2	Major SBWR Test Facilities.....	1-8
1.3	Strategy for Determination of Test and Analysis Needs.....	1-17
1.4	Overall Test and Analysis Plan.....	1-20
1.4.1	Relationship of TAPD Document to Overall TRACG Validation.....	1-20
1.4.2	List of Reports to be Submitted to the NRC.....	1-21
2.0	IDENTIFICATION OF IMPORTANT THERMAL-HYDRAULIC PHENOMENA: TOP-DOWN PROCESS.....	2-1
2.1	Introduction.....	2-1
2.2	Analysis of Events.....	2-2
2.2.1	Loss-of-Coolant Accident (LOCA).....	2-2
2.2.2	Anticipated Transients.....	2-10
2.2.3	Anticipated Transient Without Scram (ATWS).....	2-13
2.2.4	Stability.....	2-14
2.3	Phenomena Identification and Ranking Tables (PIRT).....	2-26
2.3.1	Loss-of-Coolant Accident (LOCA).....	2-27
2.3.2	Anticipated Transients.....	2-27
2.3.3	Anticipated Transient Without Scram.....	2-28
2.3.4	Stability.....	2-29
3.0	IDENTIFICATION OF SBWR-UNIQUE FEATURES AND PHENOMENA: BOTTOM-UP PROCESS.....	3-1
3.1	Introduction.....	3-1
3.2	Methodology.....	3-1
3.3	Results.....	3-3
3.3.1	RPV and Internals (B11).....	3-3
3.3.2	Nuclear Boiler System (B21).....	3-3
3.3.3	Isolation Condenser System (B32).....	3-3
3.3.4	Standby Liquid Control System (C41).....	3-3
3.3.5	Gravity-Driven Cooling System (E50).....	3-3
3.3.6	Fuel and Auxiliary Pools Cooling System (G21).....	3-3

TABLE OF CONTENTS (Continued)

3.3.7	Core (J-Series).....	3-3
3.3.8	Containment (T10).....	3-3
3.3.9	Passive Containment Cooling System (T15)	3-3
4.0	EVALUATION OF IDENTIFIED PHENOMENA AND INTERACTIONS.....	4-1
4.1	Composite List of Identified Phenomena and Interactions.....	4-1
4.2	Analytical Evaluation of System Interactions.....	4-1
4.2.1	Accident Scenario Definition	4-2
4.2.2	Results from the Primary Systems Interactions Study	4-2
4.2.3	Results from the Containment Systems Interactions Study	4-2
4.2.4	Summary of System Interaction Studies.....	4-3
4.3	Summary of Evaluations.....	4-3
4.3.1	LOCA.....	4-3
4.3.2	Transients	4-3
4.3.3	ATWS and Stability.....	4-3
5.0	MATRIX OF TESTS NEEDED FOR SBWR PERFORMANCE ANALYSIS ...	5-1
5.1	Separate Effects Tests	5-1
5.2	Component Performance Tests.....	5-1
5.3	Integral System Response Tests	5-2
5.4	Plant Operating Data.....	5-2
5.5	Summary of Test Coverage.....	5-2
6.0	INTEGRATION OF TESTS AND ANALYSIS.....	6-1
6.1	TRACG Qualification Plan	6-1
6.2	Use of Data for TRACG Model Improvement and Validation.....	6-1
7.0	SUMMARY AND CONCLUSIONS.....	7-1
8.0	REFERENCES.....	8-1

TABLE OF CONTENTS (Continued)

APPENDICES

A.	TEST AND ANALYSIS PLAN (TAP).....	A-1
A.1	Introduction.....	A-1
A.2	Test and Analysis Philosophy.....	A-1
A.3	Test and Analysis Plan.....	A-4
	Attachment A1 – Test and Analysis Document Tables-of-Contents.....	A1-1
B.	SCALING APPLICABILITY.....	B-1
B.1	Scaling Summary.....	B-1
C.	TRACG INTERACTION STUDIES.....	C-1
C.1	Introduction.....	C-1
C.2	Scenario Definition for Interaction Studies.....	C-1
C.3	Primary System Interaction Studies.....	C-2
C.4	Containment Interaction Studies.....	C-2
C.5	Summary of Interaction Studies.....	C-2

LIST OF TABLES

1.2-1	Evolution of the General Electric BWR.....	1-11
1.2-2	SBWR Features and Related Experience	1-12
1.2-3	SBWR and ABWR Analysis Methods	1-13
2.2-1	GDCS Line Break Sequence of Events.....	2-16
2.2-2	LOCA Scenario with Diesel Generators Available- Additional Systems Functional	2-18
2.2-3	LOCA Scenario with Offsite Power & Diesel Generators Available.....	2-18

APPENDICES

A.2-1	Thermal-Hydraulic Test Data Groups and Description	A-44
A.2-2	SBWR Test Documentation Submittals	A-45
A.2-3	SBWR Analysis Documentation Submittals	A-47
A.3-1	Required Thermal Hydraulic Measurements-PCC Test.....	A-48
A.3-2a	PANTHERS/PCC Steady State Performance Matrix – Steam Only Tests	A-49
A.3-2b	PANTHERS/PCC Steady State Performance Matrix – Air-Steam Mixture Tests.....	A-50
A.3-2c	PANTHERS/PCC Noncondensable – Buildup Test Matrix	A-53
A.3-2d	PANTHERS/PCC Pool Water Level Effects – Test Matrix	A-53
A.3-3	PANTHERS/PCC TRACG Qualification Points.....	A-54
A.3-4	Required Thermal Hydraulic Measurements - IC Test	A-55
A.3-5a	PANTHERS/IC Steady State Performance – Test Matrix	A-56
A.3-5b	PANTHERS/IC Startup and Operation – Test Matrix	A-56
A.3-5c	PANTHERS/IC Noncondensable Gas Effects - Test Matrix.....	A-57
A.3-5d	PANTHERS/IC Water Level Effects - Test Matrix	A-57
A.3-6	PANTHERS/IC TRACG Analysis Cases.....	A-58
A.3-7	Panda Instrumentation Summary	A-59
A.3-8	Instrumentation Required for Test S1 to S13	A-60
A.3-9a	PANDA Steady State PCC Performance Test Matrix	A-61
A.3-9b	PANDA Systems Test Matrix Summary.....	A-62
A.3-10a	Initial Conditions for PANDA Test M3	A-63
A.3-10b	Initial Conditions for PANDA Test M7	A-63
A.3-11	SBWR Containment Conditions at 3600 sec for Main Steamline Break LOCA.....	A-64
A.3-12	Time Derivatives of Key PANDA Initial Conditions	A-66
A.3-13	PANDA TRACG Analysis Cases	A-67

LIST OF TABLES (Continued)

APPENDICES (Continued)

A.3-14	GIST Test Matrix Initial Conditions (RPV at 100 psig)	A-68
A.3-15	GIST Runs With Existing TRACG Analysis	A-70
A.3-16	GIRAFFE Test Matrix (Phase 1 Step-1).....	A-71
A.3-17	GIRAFFE/Helium Integral Systems Test Matrix.....	A-72
A.3-18	GIRAFFE/Helium Base Case (H1) Initial Conditions	A-72
A.3-19	GIRAFFE/Helium "Tie-Back" Initial Conditions.....	A-73
A.3-20	GIRAFFE/Helium Test T2 Initial Conditions.....	A-74
A.3-21	GIRAFFE/SIT Test Matrix	A-74
A.3-22	Test GS1 Initial Conditions.....	A-75
A.3-23	Basis For GIRAFFE/SIT Test Conditions	A-75
A.3-24	PANTHERS PCC Structural Instrumentation	A-76
A.3-25	PANTHERS PCC Component Demonstration Test Matrix	A-77
A.3-26	PANTHERS/PCC LOCA Cycle Time History	A-77
A.3-27	PANTHERS IC Structural Measurements	A-78
A.3-28	PANTHERS IC Component Demonstration Test Matrix.....	A-79
A.3-29	Comparison of Non-Dimensional Parameters Between SBWR and CRIEPI.....	A-80

LIST OF FIGURES

1.1-1	TAPD Focus.....	1-3
1.2-1	Evolution of the BWR.....	1-14
1.2-2	Evolution of the BWR.....	1-15
1.2-3	Comparison of BWR Containments.....	1-16
1.3-1	Strategy for Determination of Test Needs.....	1-19
1.4-1	Technology Basis for SBWR Design.....	1-22
1.4-2	Overall Test and Analysis Plan.....	1-23
1.4-3	Road Map of SBWR TRACG Related Documentation.....	1-24
2.2-1	SBWR Passive Safety Systems.....	2-19
2.2-2	Phases of the LOCA Transient.....	2-20
2.2-3	GDCS Line Break-Chimney Two-Phase Level vs. Time.....	2-20
2.2-4	GDCS Line Break Containment Pressure and Temperature vs. Time.....	2-21
2.2-5	GDCS Line Break Decay Heat and PCC Power vs. Time.....	2-22
2.2-6	MSIV Closure ATWS with Boron Injection.....	2-23
2.2-7	SBWR Stability Design Criteria and Performance.....	2-25
2.2-8	SBWR Power/Flow Map Comparison With Calculated Stability Limit.....	2-25
3.2-1	SBWR Product Structure.....	3-2
6.1-1	Technology Basis for SBWR Design.....	6-2

APPENDICES

A.3-1	Passive Containment Cooler Test Article.....	A-81
A.3-2	PANTHERS/PCC Test Facility Schematic.....	A-82
A.3-3	PCC Heat Exchanger Operational Modes.....	A-83
A.3-4	Comparison of PANTHERS/PCC Steam-Air Test Range to SBWR Conditions.....	A-84
A.3-5	TRACG PANTHERS/PCC Qualification Points.....	A-85
A.3-6	Isolation Condenser Test Article.....	A-86
A.3-7	PANTHERS/IC Test Facility Process Diagram.....	A-87
A.3-8	PANDA Facility: IC/PCC Test Units.....	A-88
A.3-9	PANDA Facility Schematic.....	A-89
A.3-10	PANDA Facility: Configuration of Vessels.....	A-90
A.3-11	PANDA Facility: PCC 3 Steady State Supply Line.....	A-91
A.3-12	Deleted.....	A-92
A.3-13a	PANDA Instrumentation: Condenser, Pool, and Vessel Temperatures.....	A-93
A.3-13b	PANDA Instrumentation: Mass Flow Rates.....	A-94
A.3-13c	PANDA Instrumentation: Absolute and Differential Pressures.....	A-95
A.3-13d	PANDA Instrumentation: Noncondensable (C) and Phase Detectors.....	A-96

LIST OF FIGURES (Continued)

APPENDICES (Continued)

A.3-14	PANDA Steady State Test Instrumentation/Configuration	A-97
A.3-15	Comparison of PANDA Steady State Test Range to SBWR Condition	A-98
A.3-16	GIST Facility Schematic	A-99
A.3-17	GIST Facility Piping Arrangement	A-100
A.3-18	GIRAFFE Test Facility Schematic (Phase 1)	A-101
A.3-19	GIRAFFE IC/ PCC Unit	A-102
A.3-20	GIRAFFE Test Facility Schematic (Post Phase 1).....	A-103
A.3-21	GIRAFFE PCC Unit (Shortened Tubes)	A-104
A.3-22	PCC Startup Initial Condition Map	A-105
A.3-23	IC Cycle Types	A-106
A.3-24	CRIEPI Test Loop Outline.....	A-107
A.3-25	CRIEPI Facility Stability Map Under Lower Pressure Startup Conditions and Representative Parameters for SBWR	A-108

ABBREVIATIONS AND ACRONYMS

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

ABBREVIATIONS AND ACRONYMS (Continued)

GEXL	General Electric Critical Quality Boiling Length Correlation
GIRAFFE	Gravity-Driven Integral Full-Height Test for Passive Heat Removal
GIST	GDCS Integral System Test
HCU	Hydraulic Control Unit
HVAC	Heating, Ventilating and Air Conditioning
IC	Isolation Condenser
ICS	Isolation Condenser System
INEL	Idaho National Engineering Laboratory
LASL	Los Alamos Scientific Laboratory
LB	Large Break
LOCA	Loss-Of-Coolant Accident
LOOP	Loss Of Offsite Power
LPCI	Low Pressure Coolant Injection
MCPR	Minimum Critical Power Ratio
MIT	Massachusetts Institute of Technology
MPL	Master Parts List
MSIV	Main Steamline Isolation Valve
MSL	Main Steamline
MW	Megawatt
NBS	Nuclear Boiler System
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
P&ID	Process and Information Diagram
PANDA	Passive Nachwarmeabfuhr-und Druockabbau-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 Data base
RC&IS	Rod Control and Information System
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Cleanup
SB	Small Break

ABBREVIATIONS AND ACRONYMS (Continued)

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

1.0 INTRODUCTION

1.1 Purpose

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

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

This document describes the steps in this process leading to the final Test and Analysis Plan (Appendix A). The TRACG computer code is used for the analysis of SBWR transients, Loss-of-Coolant Accidents (LOCAs), Anticipated Transients Without Scram (ATWS) and stability. The Test Plan has been cross-referenced against the identified phenomena to create the TRACG Qualification Matrix. Section 1.3 describes in more detail the strategy employed to arrive at these objectives. The use of specific tests in the development of TRACG models, for test predictions and for post-test validation, is addressed in this report. Descriptions of the SBWR-specific test facilities and their fidelity with respect to scaling the SBWR plant are provided in Appendices A, B and Reference [32].

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 in the chimney and downcomer and peak cladding temperature (PCT)] with operation of the Emergency Core Cooling Systems (ECCS), as well as the containment pressure and temperature response to postulated breaks. Long-term core cooling by inventory makeup is also considered.

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

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

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

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

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

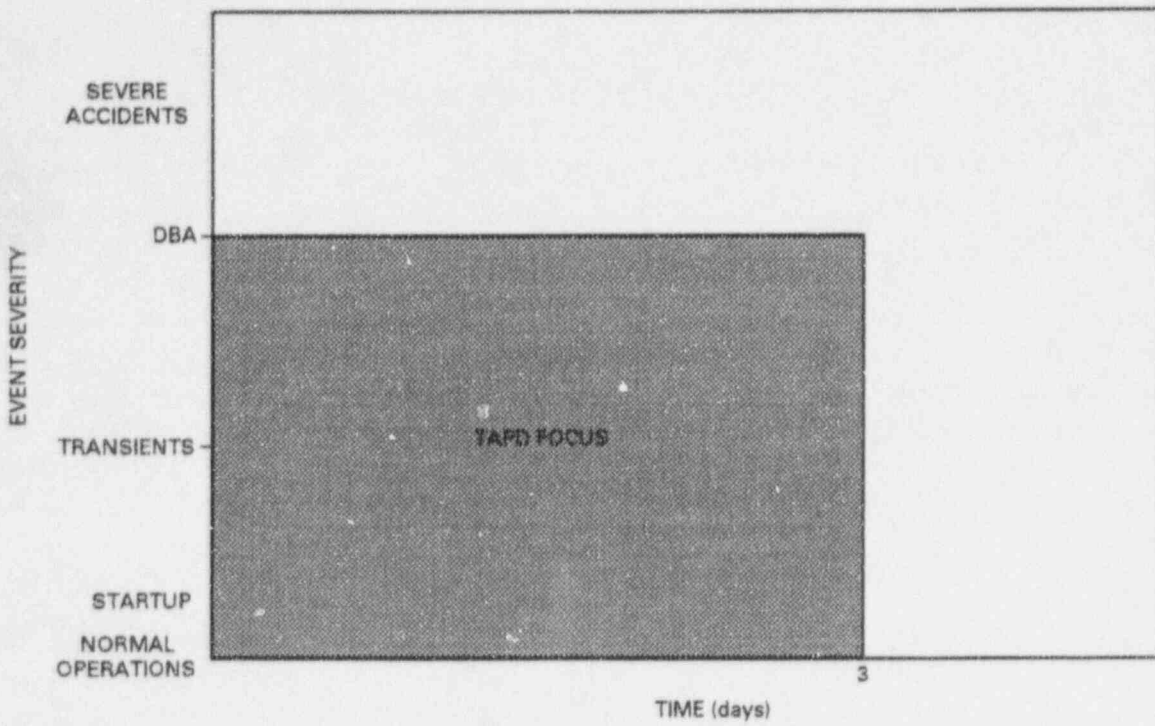


Figure 1.1-1 TAPD Focus

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). For example, the Residual Heat Removal (RHR) heat exchangers in currently operational BWRs provide an extensive experience base for the design/fabrication/testing/operation/maintenance of condensing mode heat exchangers. The SBWR Passive Containment Cooling System (PCCS) heat exchangers take advantage of this experience, are designed specifically for the SBWR requirements, and are being thoroughly tested as described in Appendix A to this report. 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 data base [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 data base 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 downcomer sensed water level.

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

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

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

1.2.1.3 Scope of Application of TRACG to SBWR

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

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

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

1.2.1.3.1 Transient Analysis

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

The analysis determines the most limiting event for the AOOs in terms of Critical Power Ratio (CPR) and margin loss (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 (Subsection 1.2.2) is used for this purpose.

1.2.2 Major SBWR Test Facilities

GE has used a procedure similar to the Code Scaling, Applicability and Uncertainty (CSAU) methodology developed by the NRC [4], [6] and submitted to the NRC by GE letter [41]. This procedure developed a list of phenomena important to the SBWR behavior in a large number of anticipated and hypothetical events and matched them against information available from operating plant and/or test experience. The Phenomena Identification and Ranking Table (PIRT) discussed in Section 2 of this report identifies specific governing phenomena, of which a significant fraction were concluded to be "important" in prediction of SBWR transient and LOCA performance. TRACG contains models capable of simulation of each of the important phenomena, and each has been qualified by the successful predictions of at least one, and in most cases, several test data sets. The PIRT defines more than 900 specific data sets, from 42 different tests and test facilities, that make up the TRACG qualification data base. 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 data base: that is, a heat transfer correlation for steam condensation in tubes in the presence of noncondensable gases. A test program has since been conducted to secure this information, reported to the NRC in Reference 19.

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

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

"tuning" of the TRACG inputs is to be performed, and no modifications to the coding are anticipated as a result of these tests.

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

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

1.2.2.1 Major SBWR-Unique Test Programs

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

1.2.2.1.1 GIST

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

1.2.2.1.2 GIRAFFE

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

1.2.2.1.3 PANDA

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

1.2.2.1.4 PANTHERS

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

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

1.2.2.1.5 Scaling of Tests

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

Table 1.2-1 Evolution of the General Electric BWR

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

Table 1.2-2 SBWR Features and Related Experience

SBWR Feature	Plants	Testing
IC	Dodewaard, Dresden 1,2,3, Big Rock Pt., Tarapur 1,2, Nine Mile Pt. 1, Oyster Creek, Millstone 1, Tsuruga, Nuclenor, Fukushima 1	Operating Plants
Natural Circulation	Dodewaard, Humboldt Bay	Operating Plants
Squib Valves (SLCS, GDCS, DPVs)	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)
Safety Relief Valves (SRVs)	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 SBWR and ABWR Analysis Methods

Analysis Type	Analysis Method	
	ABWR	SBWR
Steady state	ISCOR/RODAN	ISCOR/TRACG
Transients		
· Pressurization	OLYN/TASC	TRACG
· Loss of feedwater heating	PANACEA	PANACEA
· Other	REDY/TASC	TRACG
ATWS	REDY/TASC	TRACG
Stability	FABLE/REDY	FABLE/TRACG
LOCA/ECCS	SAFER	TRACG
LOCA/containment		
· Pressure/temperature response	M3CPT/SUPERHEX	TRACG
· Loads	Approved Methodology	Approved Methodology

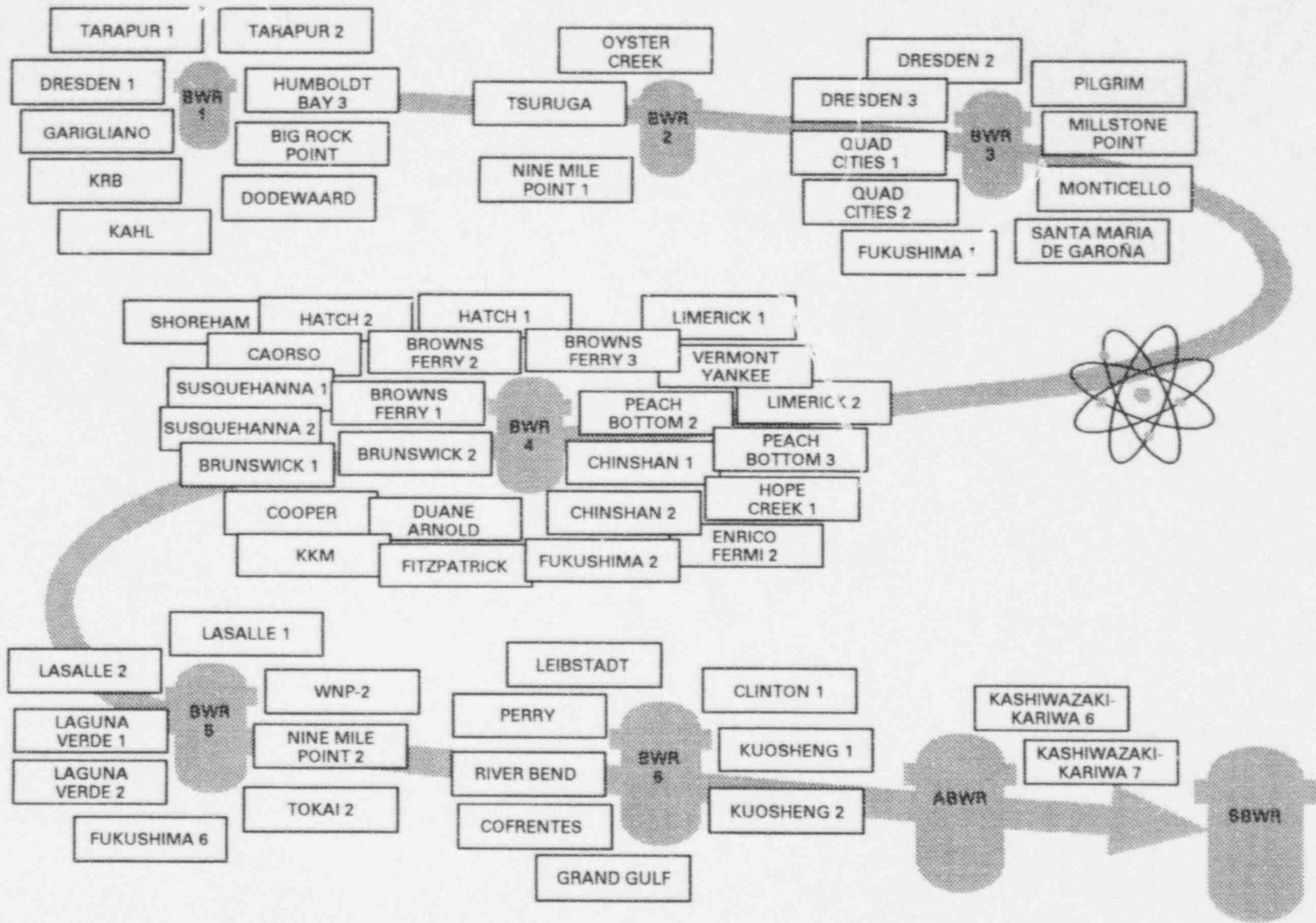


Figure 1.2-1 Evolution of the BWR

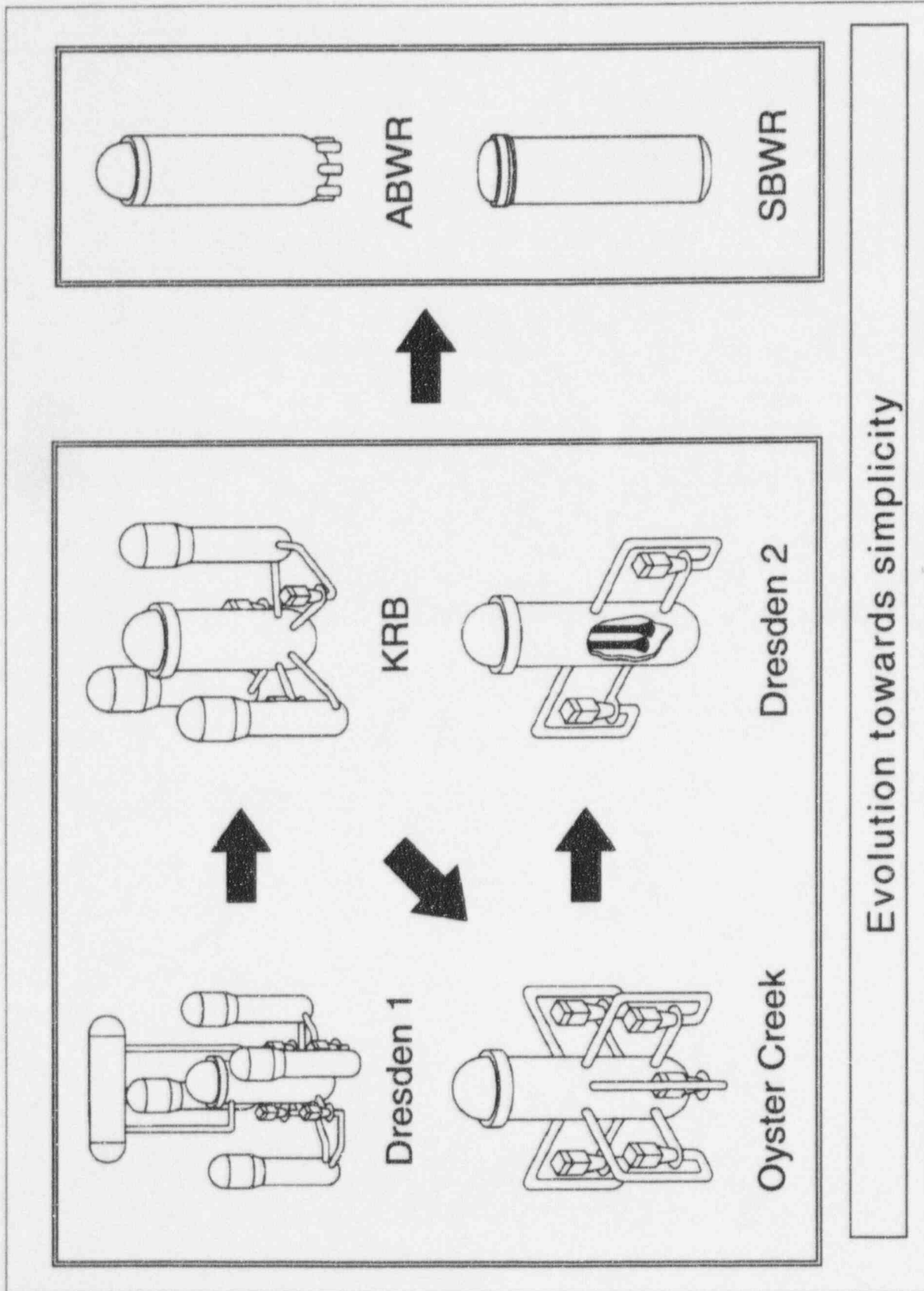


Figure 1.2-2 Evolution of the BWR

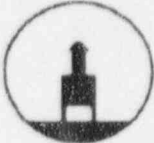





	 ISOLATION CONDENSER DRY	 CONTAINMENT FLOODER MARK I	 CONTAINMENT FLOODER MARK II	 ISOLATION CONDENSER CONTAINMENT FLOODER MARK III	 ISOLATION CONDENSER ABWR	 ISOLATION CONDENSER SBWR
PRESSURE SUPPRESSION	NO	YES	YES	YES	YES	YES
NUMBER OF BARRIERS						
CONTAINMENT	1	2	2	3	2	2
FISSION	2	4	4	4	4	4
VOLUME (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 LOC/containment scenario can be divided into (1) the *Blowdown phase*, where the reactor vessel depressurizes, enabling the Gravity-Driven Cooling System (GDCS) to start injecting water into the reactor vessel; (2) the *GDCS phase* during which the GDCS tanks drain into the reactor pressure vessel; and (3) the *Long-term cooling phase*, after the GDCS tanks have drained and the Passive Containment Cooling System (PCCS) removes decay heat and recycles condensed steam to the reactor vessel. For each phase of the transient, phenomena that might be important were listed and ranked to produce Phenomena Identification and Ranking Tables (PIRT). These tables were developed for each region of the reactor vessel and containment. This Top-Down process and the results are described in Section 2.

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

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

All the phenomena and interactions identified as important were evaluated. A Qualification Data base 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 data base. 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 data base. 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 Reference 32.

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

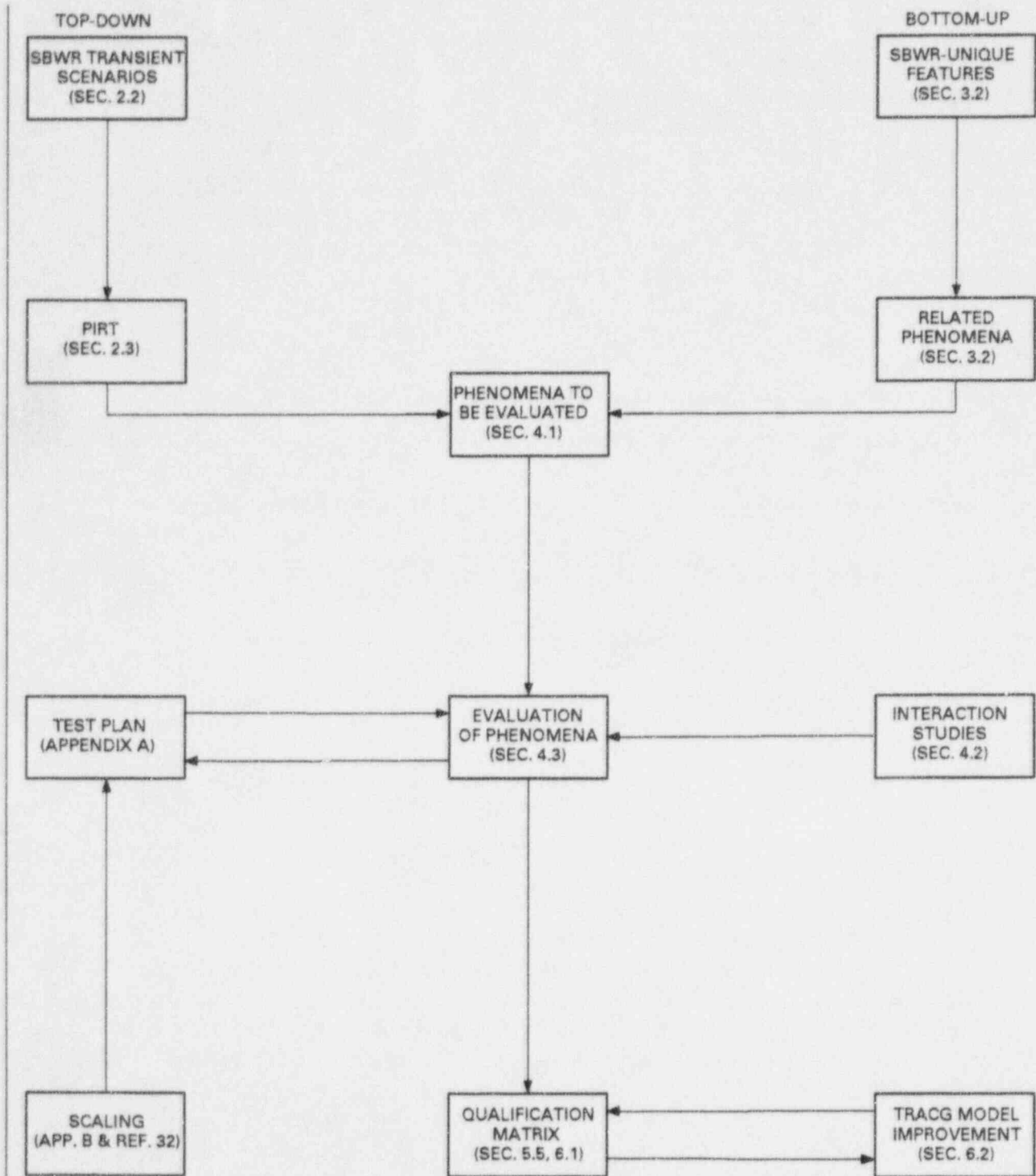


Figure 1.3-1 Strategy for Determination of Test Needs

1.4 Overall Test and Analysis Plan

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

1.4.1 Relationship of TAPD Document to Overall TRACG Validation

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

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

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

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

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

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

1.4.2 List of Reports to be Submitted to the NRC

The following is a list of Licensing Topical Reports planned to be submitted. (See Appendix A, Attachment 1.

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

In addition, there will be a Licensing Topical Report covering the SBWR Test Program..

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

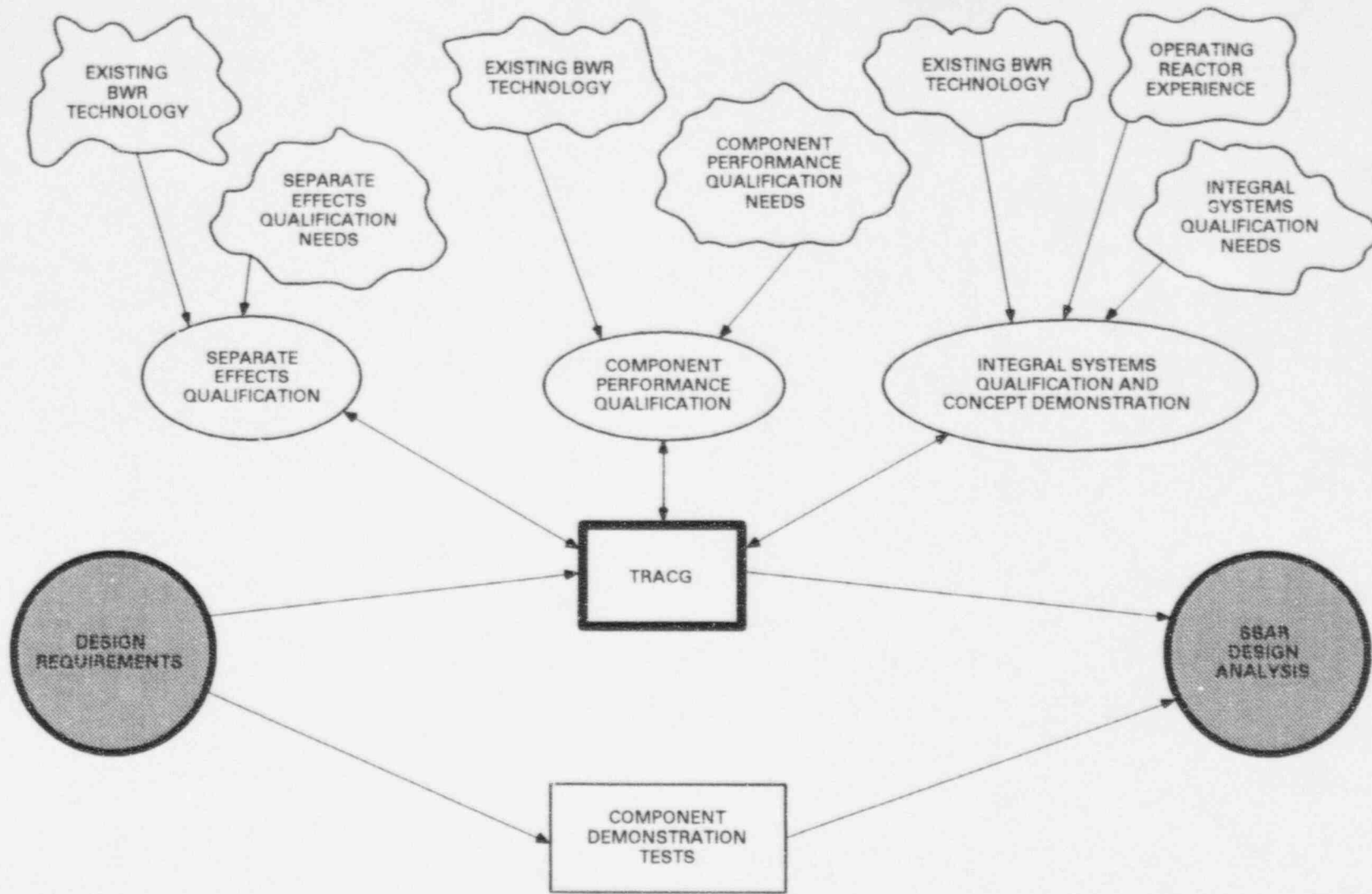


Figure 1.4-1 Technology Basis for SBWR Design

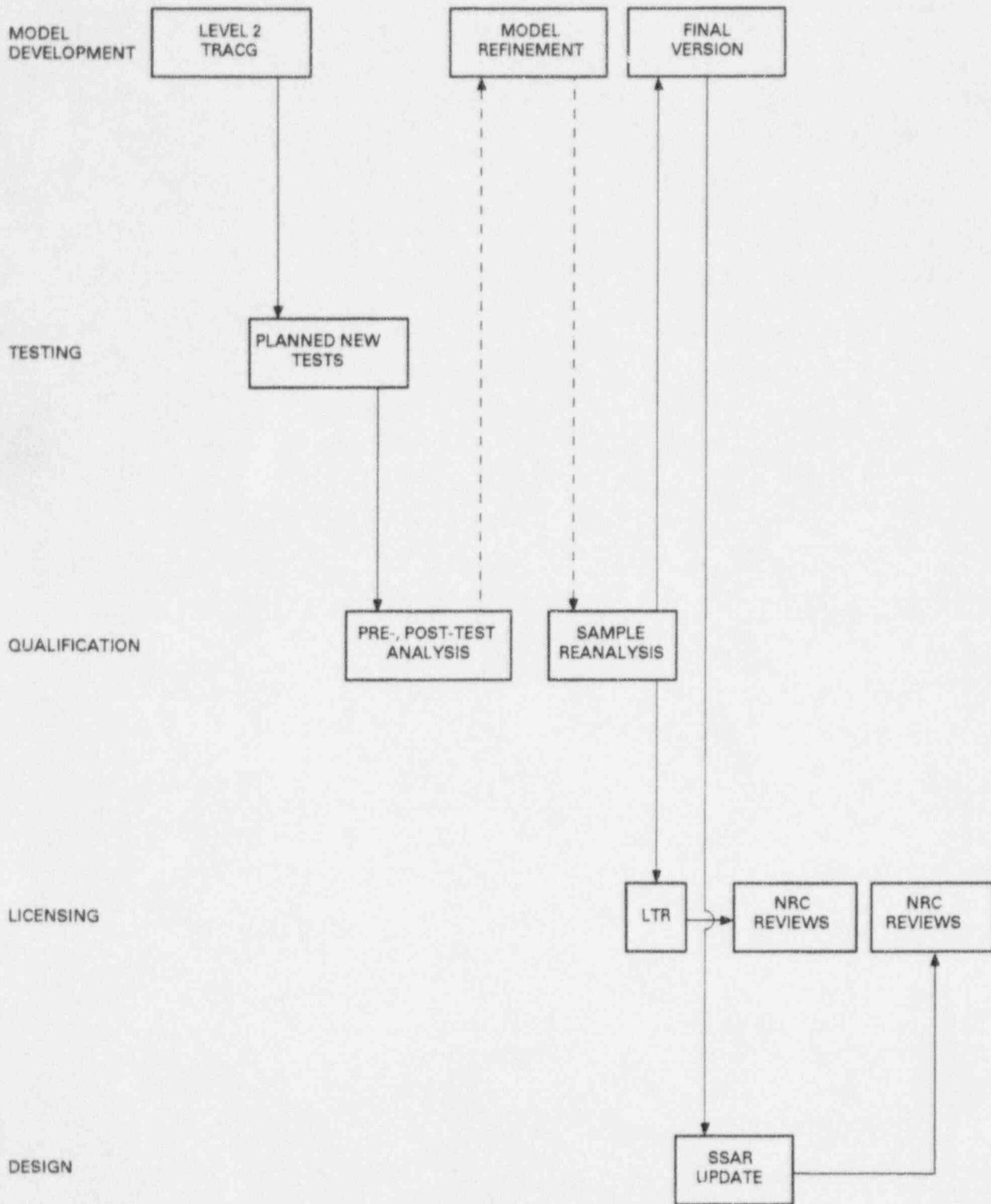


Figure 1.4-2 Overall Test and Analysis Plan

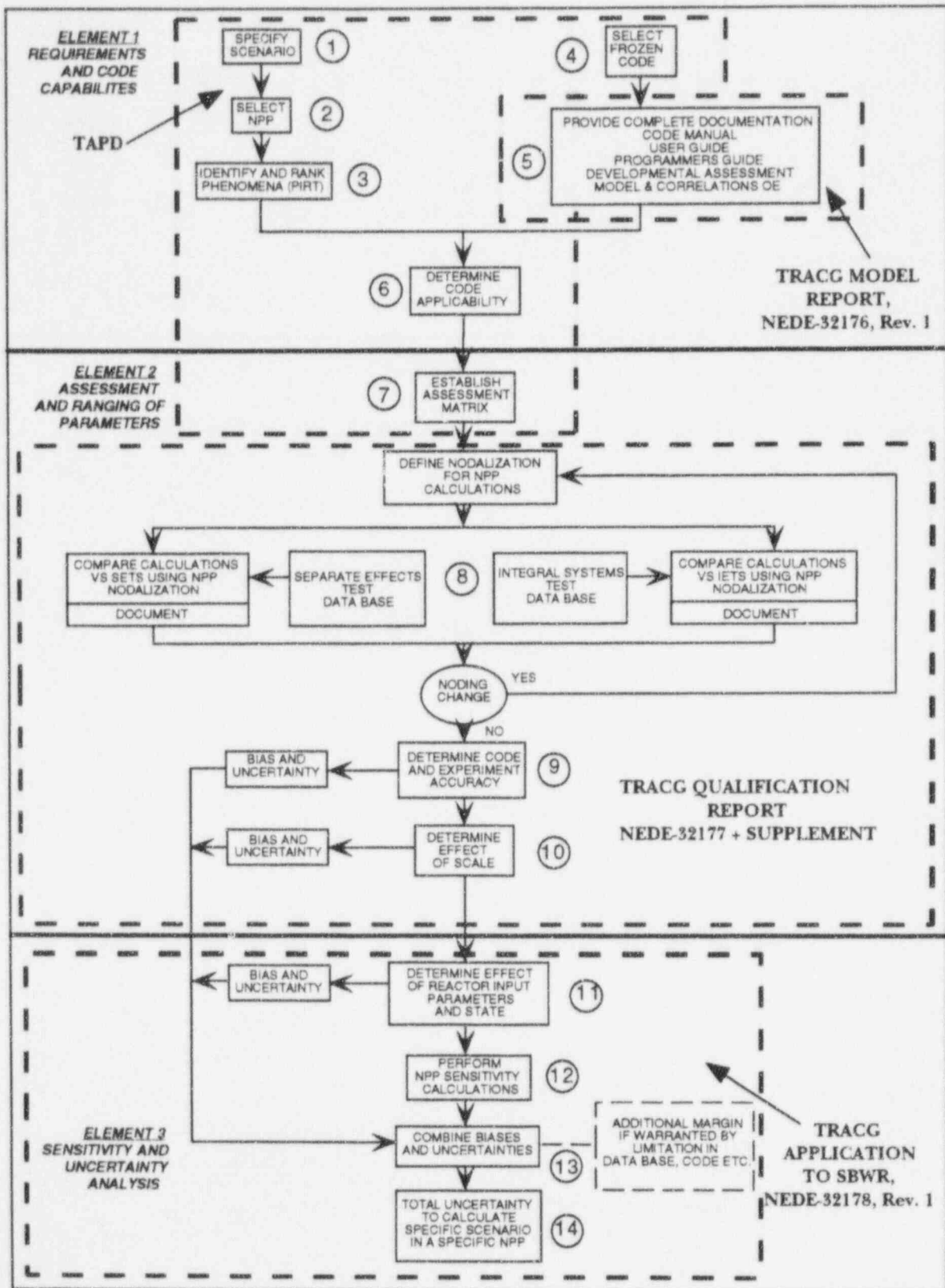


Figure 1.4-3 Road Map of SBWR TRACG Related Documentation

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

2.1 Introduction

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

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

The analyses were then reviewed by interdisciplinary teams to identify each thermal-hydraulic phenomenon that plays a role in the analysis, and to rank all of them in terms of "importance"; that is, degree of influence on some figure of merit (e.g., two-phase level inside the shroud, containment pressure). NEDC-32391P Supplement 1 has a more detailed description of the PIRT parameters and the rationale for their relative importance. Both Section 2.3 and NEDC-32391P Supplement 1 refer back to the scenarios developed in Section 2.2. The organization structure of these sections is summarized below:

Event Categories	Analysis of Events	PIRT Summary	PIRT Detailed Description
LOCA/ECCS	Section 2.2.1.1	Section 2.3.1.1	Supl. 1 Sec. S1.3.1
LOCA/Containment	Section 2.2.1.2	Section 2.3.1.2	Supl. 1 Sec. S1.3.2
Transients	Section 2.2.2	Section 2.3.2	Supl. 1 Sec. S1.3.3
ATWS	Section 2.2.3	Section 2.3.3	Supl. 1 Sec. S1.3.4
Stability	Section 2.2.4	Section 2.3.4	Supl. 1 Sec. S1.3.5

2.2 Analysis of Events

2.2.1 Loss-Of-Coolant Accident (LOCA)

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

The overall LOCA sequence can be divided into three periods: blowdown period, GDCS period and the long-term cooling PCCS period. These periods are shown in Figure 2.2-2. The **Blowdown period** is characterized by a rapid depressurization of the vessel through the break, safety relief valves (SRVs) and depressurization valves (DPVs). The steam blowdown from the break and DPVs pressurizes the drywell, clearing the main containment vents and the PCCS vents. First, 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 elevation 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 inventory 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 (CACCS) 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 measured water level in the downcomer to drop past the Level 3 (L3) scram setpoint. The "measured" or "sensed" downcomer level corresponds to the static head in the downcomer above the lower instrument tap used for the wide range level instrument. This setpoint is assumed to scram the reactor. The scram will temporarily increase the rate of measured downcomer level drop and the Level 2 (L2) trip will quickly follow the L3 trip. This trip will isolate the steamlines and open the isolation condenser (IC) drain valves, but no credit is taken in the safety analysis for heat removal by the IC. After L2, the rate of decrease in the downcomer sensed level 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 downcomer level swell. Similarly, opening of the depressurization valves (DPVs) is delayed 45 seconds. Two DPVs on the main steamlines 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 downcomer and chimney, 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 two-phase level rises. When the two-phase 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 two-phase level above the break, until the pools empty, then the level drains back to the break elevation. Inside the core shroud, the two-phase level in the chimney also decreases after depressurization, but is restored after the GDCS refills the vessel. Figure 2.2-3 shows the chimney two-phase level during the first 25 minutes of the transient. The two-phase 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 two-phase level falls below the GDCS line elevation, the break becomes a vapor blowdown. The ADS, activated by the measured 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. Within the pool, temperature stratification occurs, with the blowdown energy being absorbed primarily in the region above the open vents. The increase in drywell pressure establishes flow through the PCCS, which also absorbs part of the blowdown energy. For the GDCS break, this period of the accident lasts less than 10 minutes. The peak containment pressure in the short term is primarily set by the compression of the noncondensibles initially in the drywell into the wetwell vapor space. The controlling parameters are the ratio of the drywell to wetwell vapor volumes, and the temperature at the top of the suppression pool, which sets the steam partial pressure.

- **GDCS Period** — Once the vessel pressure drops below the setpoint of the check valves in the two unbroken GDCS lines, the GDCS pools begin to empty their inventory into the vessel. The subcooled GDCS water quenches the core voids, stopping the steam flow from the vessel. The GDCS flow refills the vessel to the elevation 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 re-established 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. Any uncondensed steam condenses and deposits its latent heat in the portion of the suppression pool above the outlet of the PCCS vent. This period of the accident is expected to last approximately 3 hours for the GDCS line break.
- **Long-Term PCCS Period** — After the drywell pressure transient initiated by the GDCS flow is over, the drywell pressure settles out, slightly above the wetwell airspace pressure. A drywell-to-wetwell pressure difference is established which is sufficient to open the PCCS vent and drive the steam generated by decay heat through the PCCS. The drywell pressure and temperature during the first 12 hours of the GDCS line break transient are shown in Figure 2.2-4. The drywell pressure rises rapidly during the blowdown period, decreases at GDCS initiation, drops as the GDCS spills into the drywell and finally levels off as boiloff resumes. The temperature shown is for a node high in the drywell. At this location, the temperature rises during blowdown, then actually superheats during the GDCS period, but levels off as flow to the PCCS resumes. In lower regions of the drywell, affected by GDCS spill, the temperature may drop during the GDCS period. Figure 2.2-5 shows the PCCS power during the first 12 hours of the transient. Also shown is the decay heat. During the blowdown period, the PCCS picks up part of the energy released during the blowdown, most of which is deposited in the suppression pool. During the GDCS period, steam flow to the PCCS stops and the PCCS power drops to zero. As soon as the decay heat can overcome the GDCS subcooling, boiloff and steam flow to the PCCS resumes and by between 3 and 4 hours, the PCCS power increases back to nearly equal to the decay heat power.

By way of comparison, the drywell pressure at the beginning of the long-term period for the GDCS line break is below the drywell pressure for the large steamline break. During the 72 hours which defines the long-term cooling period, the drywell pressure remains below the large steamline 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 downcomer 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 downcomer 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 water 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 water 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 occurrences listed as "symptoms" in the first column result in "actions", which are the corresponding entries in the second column. 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 the two-phase level inside the core shroud (e.g., decay heat, energy release from heat slabs) are comparatively important. For the containment, after the blowdown and release of energy to the suppression pool, the effectiveness of the PCCS controls the containment response, with no pumped decay-heat removal system available. In the long-term cooling period, the containment pressure and temperature increase slowly until the end of the 72-hour period, at which time credit for non-safety decay-heat removal systems is permitted. Thus, containment pressure and temperature become the primary figures of merit for the containment and the phenomena affecting them are important.

The LOCA scenario develops slowly for the SBWR. The accident detection system logic functions almost instantaneously, but thereafter, the time scales are measured in hours rather than seconds. The chimney two-phase level (Figure 2.2-3) dips briefly about 10 minutes into the LOCA due to void collapse following GDCS injection. For the GDCS line break, the minimum chimney level occurs at about 7 hours after the break. At this point in time, the core void fraction is very small, and the chimney and downcomer levels are almost the same. This slow response, which is due to the large volume of water in the reactor vessel and GDCS pools,

makes the LOCA a very slow moving event from the reactor systems and operator response standpoint. Similarly, containment response (Figure 2.2-4) is gradual, not reaching the design pressure even 72 hours after the break. This slow response permits well-considered, deliberate operator actions.

2.2.1.4 Main Steamline Break

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

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

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

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

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

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

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

2.2.1.5 Small Breaks

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

2.2.1.6 Non-Design Basis LOCAs

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

Single Failures:

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

Isolation Condenser Operation:

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

Diesel Generators Available:

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

Offsite Power Available:

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

2.2.2 Anticipated Transients

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

2.2.2.1 Fast Pressurization Events

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

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

2.2.2.2 Slow Pressurization Events

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

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

2.2.2.3 Decrease in Reactor Coolant Inventory

Loss of feedwater flow is characteristic of this category of transient. The IC maintains downcomer water level. Reactor water level in the downcomer 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 downcomer 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 downcomer 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 downcomer water level is restored. Without feedwater, the downcomer 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 downcomer water level near the L2 setpoint. CRD high pressure injection will cause the downcomer water 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.

2.2.3.1 MSIV Closure Transient

The incident is initiated by the inadvertent closure of all MSIVs which isolates the reactor vessel. If the control rod scram fails, the rapid increase of vessel pressure together with the APRM not-downscale signal will generate an ATWS signal. This signal initiates the feedwater runback and activates Alternative Rod Injection (ARI), FMCRD run-in and the boron injection timer. If the alternate rod insertion fails, the squib valves in the standby liquid control system (SLCS) will blow open after the boron injection timer runs out (180 seconds). The sodium pentaborate solution is then released into the core bypass region. The highly enriched sodium pentaborate solution quickly mixes with the core coolant and achieves hot shutdown in less than 60 seconds.

The reactor pressure is limited to 10.2 MPa by the discharge of steam to the suppression pools through the safety valves as shown in Figure 2.2-6 A. The reactor thermal power (fuel rod surface heat flux) peaks at about 7 seconds, and is reduced by the feedwater runback. Power reduction and safety valve discharge cause the vessel pressure to drop. At about 100 seconds, all the safety valves reclose. Four of the safety valves will open and close for another 9 cycles until the reactor achieves hot shutdown. The steam generated by the decay heat is then removed by the IC alone (capacity 4.5% of rated power), and no more steam is discharged into the suppression pool. This limits the suppression pool temperature to 329° K with an associated containment pressure of 0.121 MPa.

The downcomer water level decreases rapidly after the initiation of feedwater runback as shown in Figure 2.2-6 B. This in turn reduces the core flow rate and the reactor power as shown in Figure 2.2-6 C. The downcomer water level keeps decreasing since the water inventory is removed through steam discharge into the suppression pool. The only make-up water is from the CRD flow which starts at about 80 seconds and accounts for only about 2% of the rated feedwater flow. The water downcomer level finally starts to recover when the reactor reaches hot shutdown. The steam will recycle through IC and the CRD flow provides the extra inventory as shown in Figure 2.2-6 D.

The rapid pressurization of the reactor vessel collapses the core voids which results in a neutron flux surge to 330% of rated condition at 2 seconds. The increase of fuel temperature and core void limit the maximum value of the core power. The reactor is finally brought down by the reduced core flow and increased voids as shown in Figure 2.2-6 C. The thermal power of the reactor reaches its peak of 140% at about 19 seconds. The power then decreases with the reduction of the core flow and settles at about 25% of rated condition as the core flow is running at about 4% rated. This reduced core flow is the result of lower downcomer water level and reactor power which provides the buoyancy to drive the core flow. After attaining hot shutdown from boron injection, the power follows the decay heat generation rate.

The initial surge of power creates conditions for the boiling transition in some of the hot rods at the high powered channels. The peak cladding temperature reaches 408°C at 21 seconds which is well below the design limits of 1200°C and fuel integrity is not compromised.

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 2.2-7). The core decay ratio is maintained less than 0.4 and the channel decay ratio less than 0.3.

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

2.2.4.1 For Steady State Operation

During *steady state operation*, the highest power/flow ratio occurs at 104.2% power and 100% flow conditions. The decay ratio is well within the conservative design criteria (Figure 2.2-7). At reduced power level, the power/flow ratio is lower, so the decay ratios for both core and hot channel are lower than at the rated condition. This conclusion is supported by Dodewaard test data as shown in the figure. The decay ratios during normal operation at Dodewaard have been very low, with no indication of any incipient instability throughout its long operating history. In Figure 2.2-8, the power/flow map of SBWR normal operation is compared with the stability limit calculated in the Oak Ridge National Laboratory (ORNL) study. The results confirm that there is large margin for stability. This indicates that the SBWR is very stable under normal operation conditions.

2.2.4.2 For Anticipated Transients

Of the *anticipated transients*, the loss of 55.6°C (100°F) feedwater heating case gives the highest power/flow ratio. Loss of feedwater flow is another limiting event. However, the scram quickly mitigates the transient and the power conditions are reduced to hot shutdown. For both events, the decay ratios for core and hot channel meet the design criteria shown in Figure 2.2-7. In Figure 2.2-8, both of these transient events are seen to result in power/flow conditions that are well below the exclusion region.

2.2.4.3 For ATWS Conditions

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

2.2.4.4 For Startup

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

Table 2.2-1 GDCS Line Break Sequence of Events

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

Notes To Table 2.2-1:

- (1) Scram on high drywell pressure and level decrease to L2 occur within one minute of the line break.
- (2) PARS will actuate as soon as they are exposed to radiolytic hydrogen, estimated to occur within a few minutes of the line break.
- (3) Increase of DW level to the spillover holes only occurs if it is assumed that inward flow through the break cannot occur. Otherwise, the inventory spilled to the DW returns to the RPV through the break.

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

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

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

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

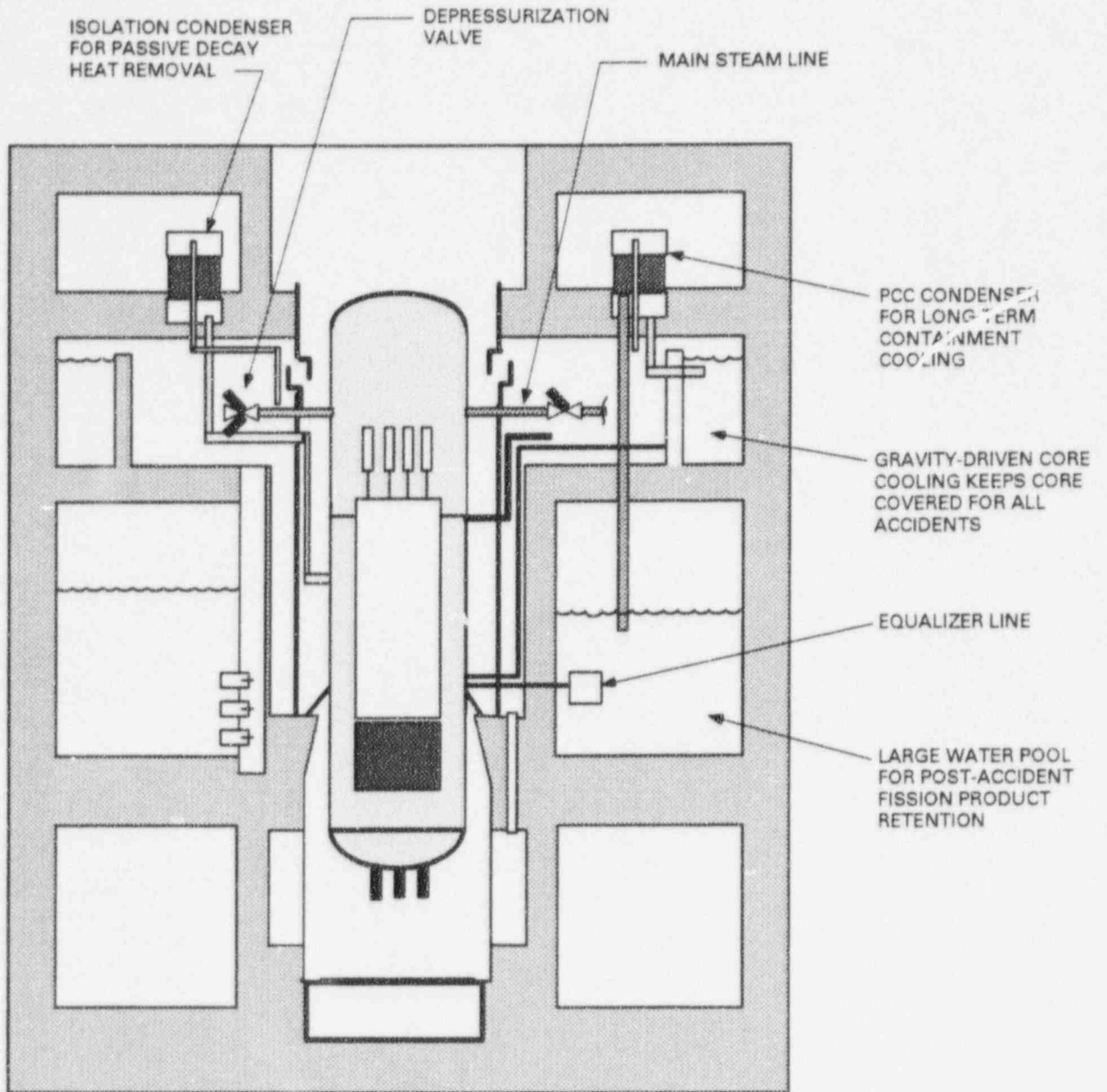


Figure 2.2-1 SBWR Passive Safety Systems

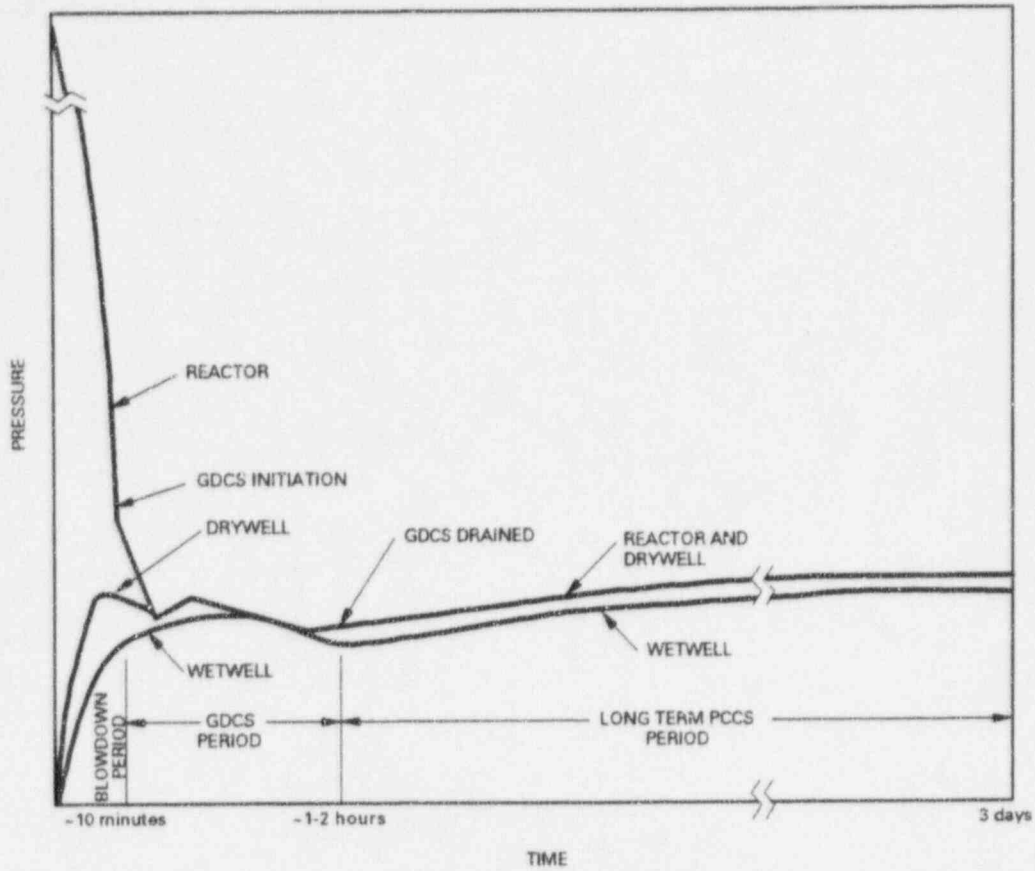


Figure 2.2-2 Phases of the LOCA Transient

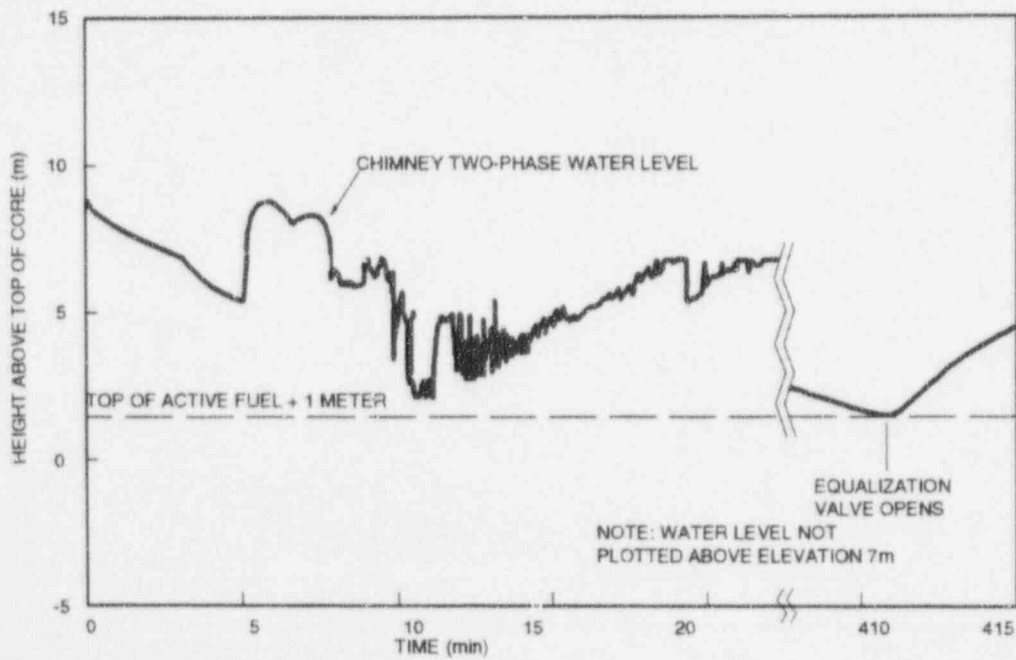


Figure 2.2-3 GDCS Line Break - Chimney Two-Phase Level vs. Time

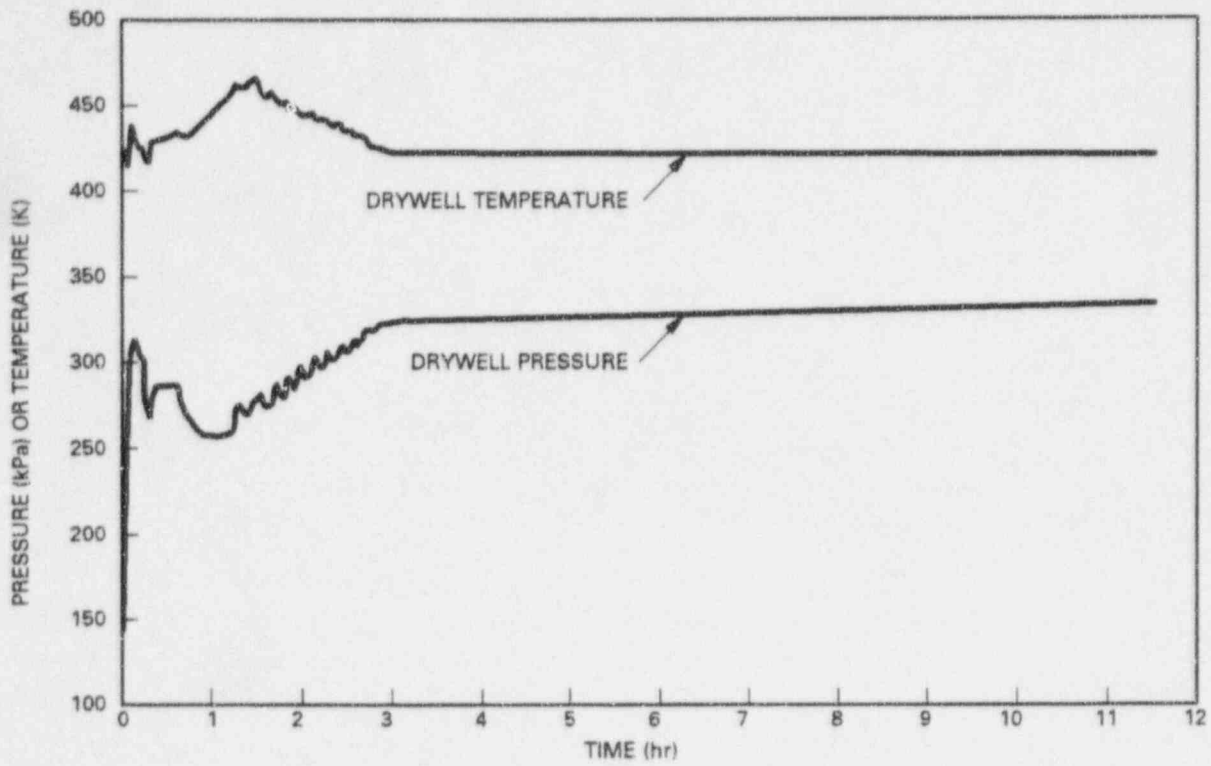


Figure 2.2-4 GDCS Line Break Containment Pressure and Temperature vs. Time

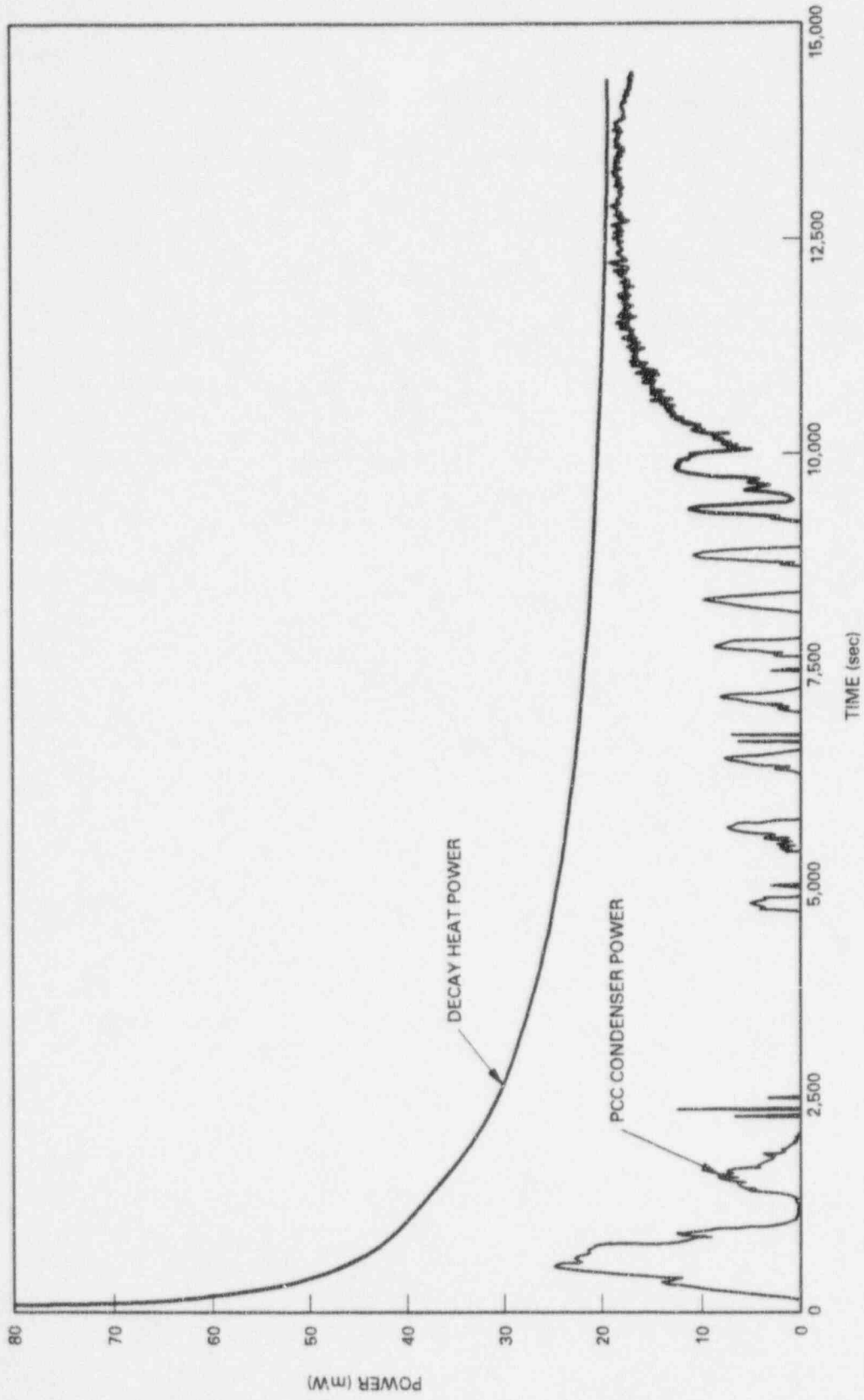
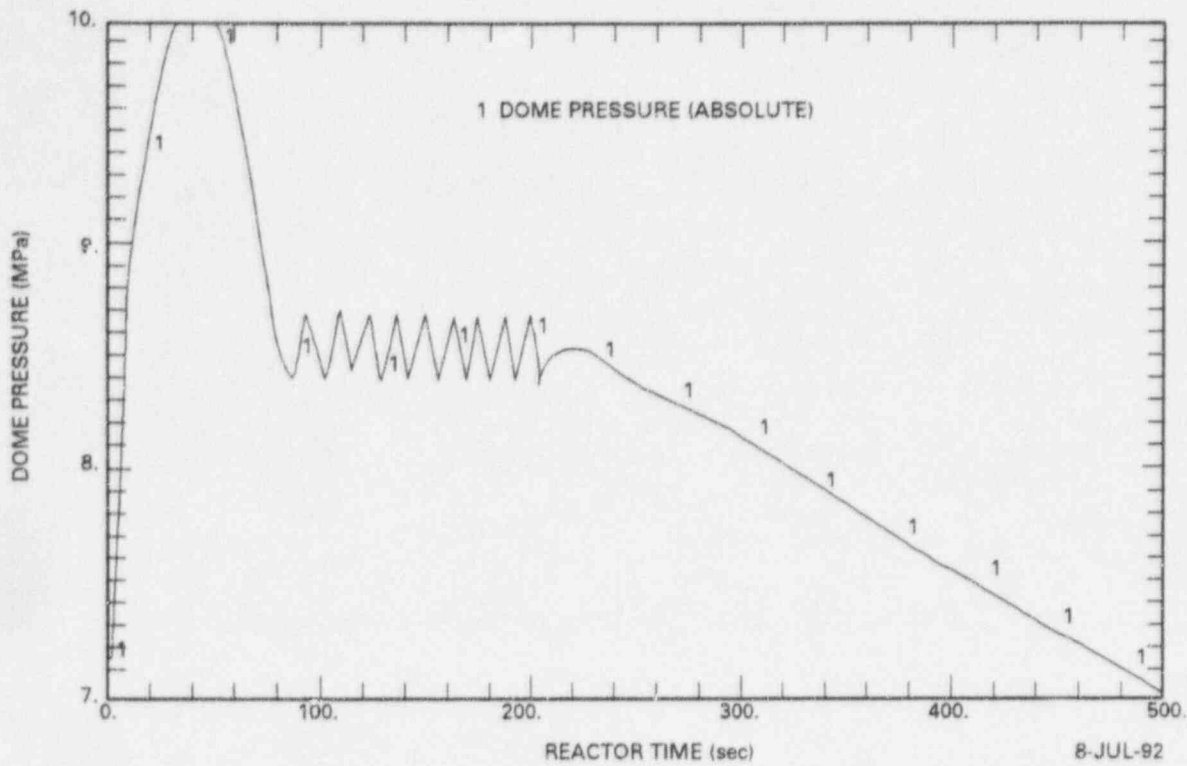
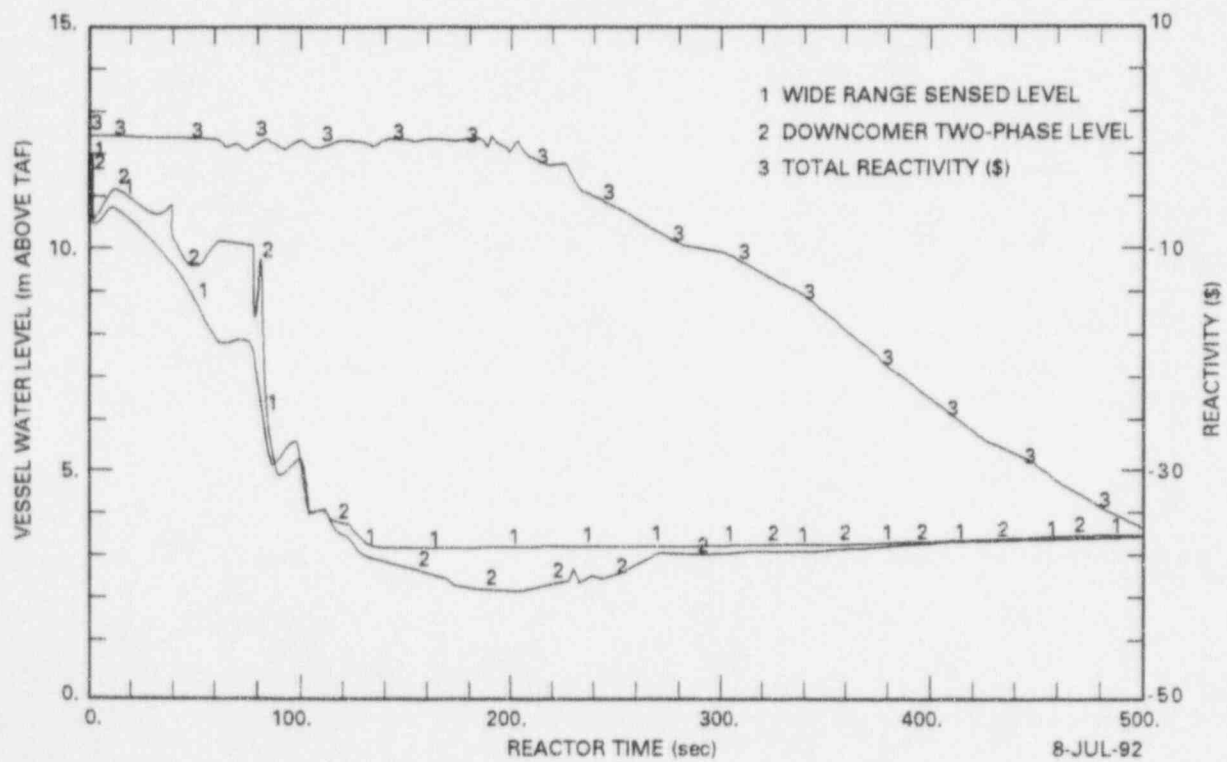


Figure 2.2-5 GDCS Line Break Decay Heat and PCC Power vs. Time

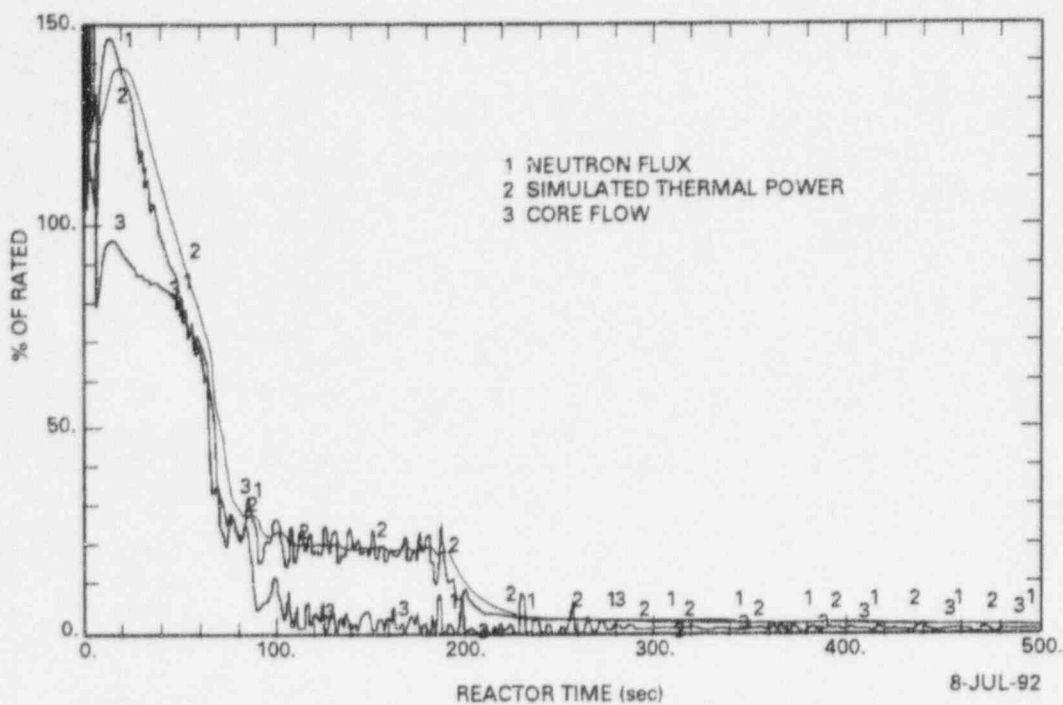


Part A

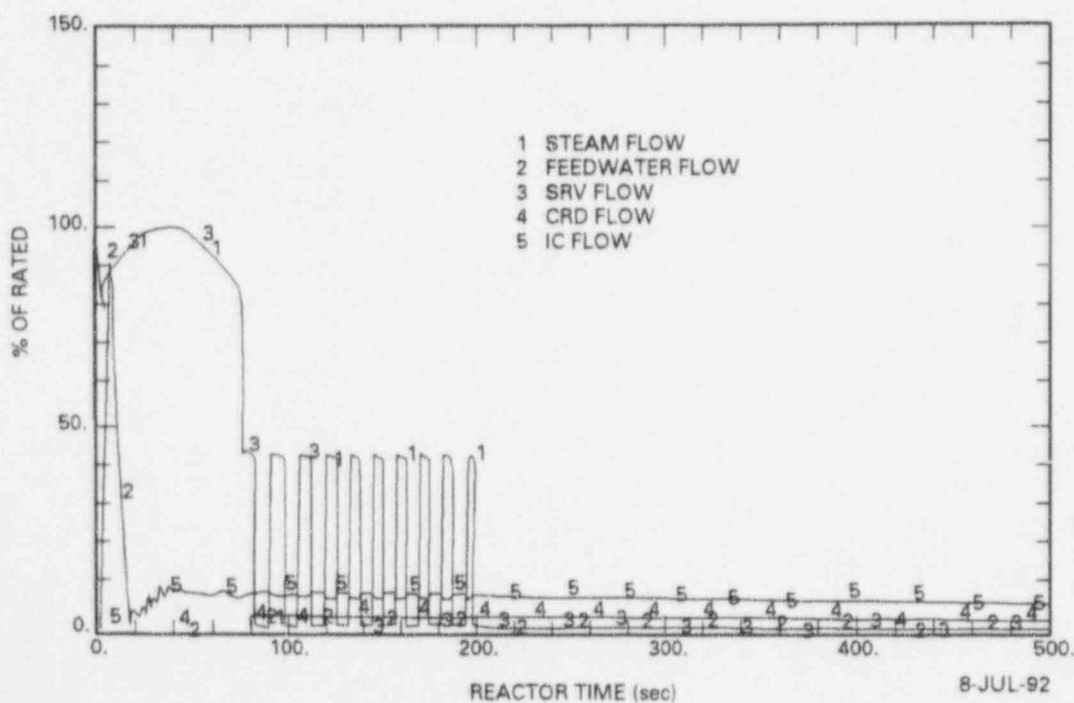


Part B

Figure 2.2-6 MSIV Closure ATWS with Boron Injection



Part C



Part D

Figure 2.2-6 MSIV Closure ATWS with Boron Injected (Continued)

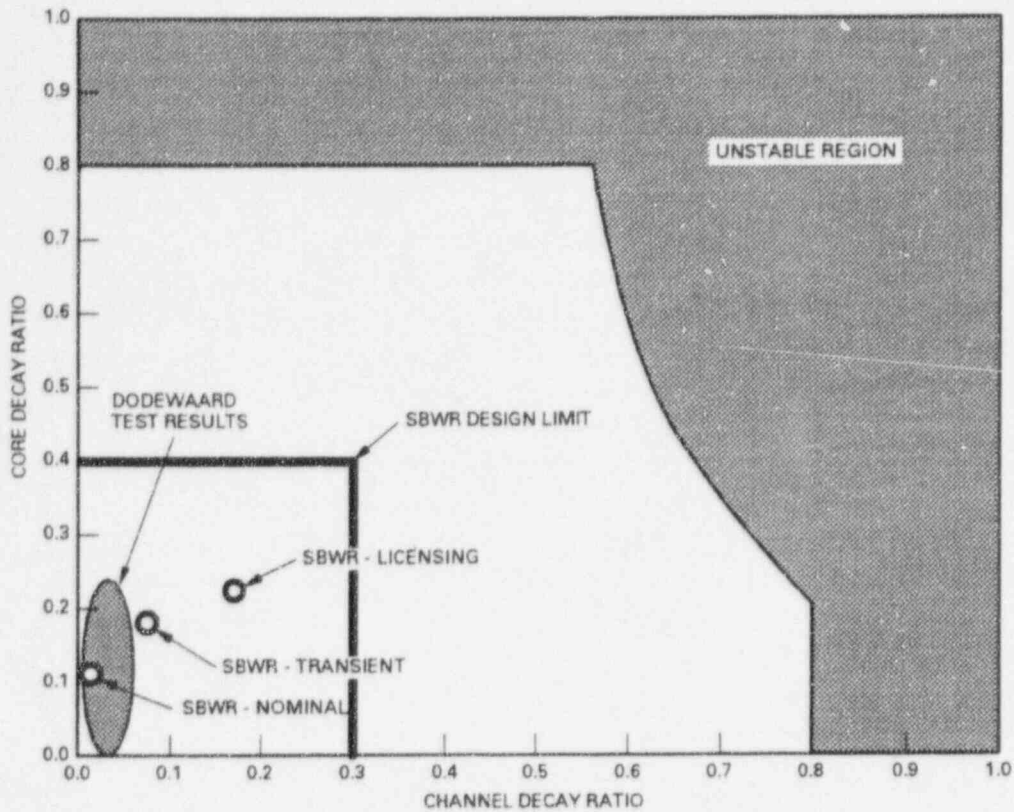


Figure 2.2-7 SBWR Stability Design Criteria and Performance

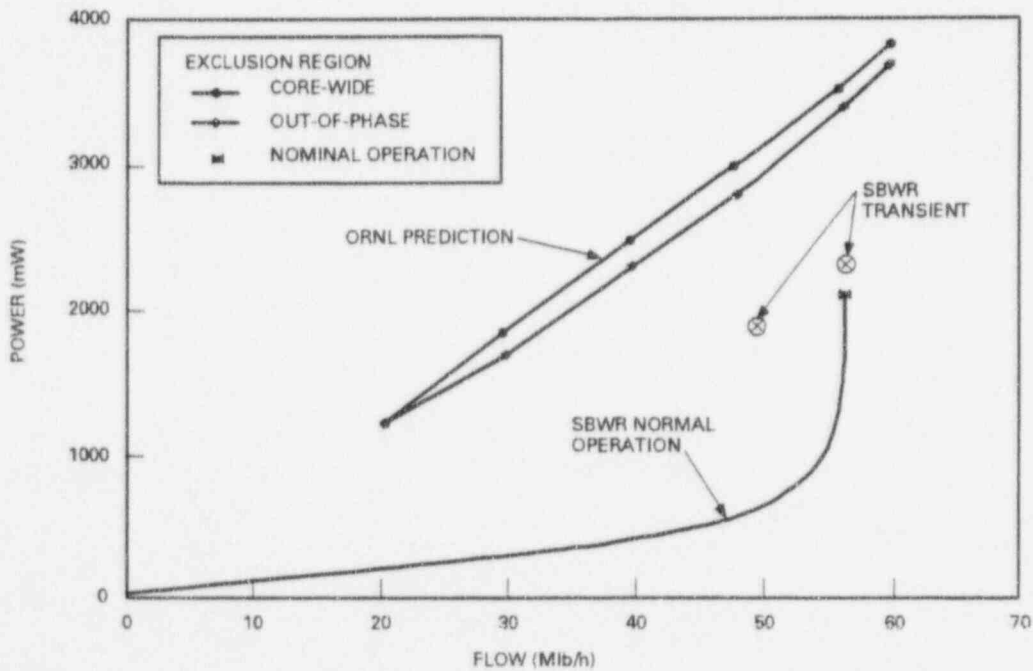


Figure 2.2-8 SBWR Power/Flow Map Comparison With Calculated Stability Limit

2.3 Phenomena Identification and Ranking Tables (PIRT)

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

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

The PIRT tables are used for three purposes.

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

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

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

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

2.3.1 Loss-Of-Coolant Accident (LOCA)

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

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

2.3.2 Anticipated Transients

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

The PIRT for transients has been revised. The discussion has been focused on specific transients. Clearer definition of the phenomena and re-evaluation of phenomena has resulted in some changes to the relative rankings. A description of the phenomena and a rationale for the importance can be found in Supplement 1 to this report. In the PIRT for transients, the pressurization transients considered are the turbine trip or load rejection without the opening of the bypass, and the inadvertent closure of the Main Steamline Isolation Valves (MSIVs). The cold water event is Loss of Feedwater Heating; inadvertent actuation of the RWCU/SDS and ICS are much milder in comparison. The Feedwater Controller Failure High is a cold water event, followed by a turbine trip. It is bounded by the two above event categories. Depressurization transients are less limiting. The most severe event in this category is the Pressure Regulator Failure Downscale, in which all turbine control valves and bypass valves are assumed to fail open. Parameters of importance to all these transients can be grouped under the following categories:

Parameters affecting initial operating state. This includes the initial natural circulation flow rate, the flow distribution among channels, power and void distribution, control rod distribution and separator carryunder.

Transient thermal hydraulic/neutronic response, characterized by pressure response, void fraction changes in the core, reactivity feedback from voids, Doppler broadening and scram, and the resulting transient core power and flow response.

Margin to boiling transition, which is determined by the fuel rod thermal response and the relationship between the core thermal hydraulic conditions and critical heat flux.

After the reactor has been scrammed, there is a long term phase of the transient that involves inventory control and bringing the reactor to a hot standby condition. This process does not involve challenging thermal hydraulic phenomena and is not considered here. This phase of the transient does not affect the MCPR and peak pressure, which are the figures of merit for these transients.

Issues related to the performance of the Isolation Condenser (IC) are considered in Section 3. Only the effects of the cold water injection in the downcomer have been included in the PIRT discussions in Section 2. Because of the neutronic coupling, the core region is by far the most important for the operational transients. Steamline dynamics play an important role in determining the pressurization rate and void collapse following a turbine trip or load rejection. The key parameters for a depressurization event are core void fraction, flashing, and void reactivity feedback. The chimney void fraction and the separator carryunder characteristics are important in determining the initial operating state.

The plant startup transient is not a MCPR limiting transient. The concern here is the margin to large oscillations in flow at low power. The conditions most likely to produce oscillations in flow are at the incipience of void generation at the top of the core and in the chimney. The flow in the core is single phase and there are no significant voids in the core. Thus, void reactivity feedback is not important unless the oscillations become very large. The plant startup transient is influenced by core inlet subcooling, rate of heatup, subcooled boiling, and the chimney subcooling and void formation.

In NEDC-32391P Table 2.3-3, the PIRT parameters for transients are listed by the region of the reactor vessel. Historically, the LOCA tables were developed first. The list of parameters considered for LOCA formed the starting point for the transient PIRT. Thus, there are some parameters that are not relevant for transients, which have been retained in the complete list. The individual items in NEDC-32391P Table 2.3-3 are discussed in NEDC-32391P Supplement 1. A total of 82 phenomena were considered and 24 were evaluated as having "High" importance for at least one of the transients. Another 15 are in the "Medium" category and the rest were ranked "Low".

2.3.3 Anticipated Transients Without Scram

The PIRT for a pressurization event (MSIV closure, turbine trip) with failure to scram is shown in NEDC-32391P Table 2.3-4. Because the event is initiated as a normal pressurization event, a large number of phenomena typical of the early phase of the transient are the same as those in the first column of NEDC-32391P Table 2.3-3 for operational transients. NEDC-32391P Supplement 1 has more discussion of the PIRT rankings.

Anticipated transients without scram progress through three phases to shutdown. The first phase is the initial transient resulting from the initiating event, for example inadvertent closure of the MSIVs. This phase is similar to the operational transients and the same phenomena are important as for the corresponding operational transient with scram. The response is more

severe because of the lack of an early scram., and boiling transition will occur. In the second phase, the lack of scram is sensed and mitigation actions are taken. These include feedwater runback to reduce flow and power, and the initiation of the boron timer. Core power is reduced to about 25 % of rated and energy removed by the periodic opening of the SRVs, discharging steam into the suppression pool. The parameters of most interest in this phase are the thermal hydraulic and neutronic interactions which lead to power reduction as the natural circulation flow drops, and the energy deposition in the suppression pool. In the final phase, boron is injected, 3 minutes after the start of the boron timer. Boron is injected into the core bypass region, mixes with the water in the bypass, moves into the core and results in hot shutdown. After the fission process has been shut down, decay heat is removed by the IC and further SRV openings are not expected. The important phenomena in this period relate to the processes governing the delivery of the boron to the core.

2.3.4 Stability

Section 2.2.4 describes the conditions for which SBWR stability is evaluated: steady state operation, anticipated transients, ATWS, and startup. Of these, the startup transient has been considered as part of the PIRT for transients. The phenomena of importance for steady state and anticipated transients are the same. The differences lie in the reactor operating conditions at which the evaluations are conducted. For example, the limiting conditions of the highest power/flow ratio are obtained at the end of the loss of feedwater heating transient. Large margins to instability are calculated for all operational transients. ATWS events lead to the most severe conditions for stability (power/flow ratio) and also to situations where the critical power may be exceeded. However, no power oscillations are expected as margins to instability are maintained, and the power/flow ratio becomes more favorable following feedwater runback. Aspects of film boiling are treated in the ATWS PIRT. For stability, phenomena affecting decay ratio and the likelihood of oscillations are considered in NEDC-32391P Table 2.3-5. Stability of a plant is significantly affected by the plant operational state: power/flow ratio, control rod distribution, and axial and radial power distributions. These have not been included in the PIRT list of phenomena. Core stability is also determined largely by the fuel design: void coefficient, two-phase/single phase pressure drop, and fuel rod time constant for heat transfer. These parameters have been included in the PIRT. A separate table of the operational and design parameters that govern stability is provided in NEDC-32391P Table 2.3-6. It should be noted that TRACG will only be used for the evaluation of ATWS and the potential for instabilities during an ATWS event. NRC approved methodology (FABLE code) is being applied for stability evaluations under steady state conditions and for operating states resulting from anticipated transients.

The BWR stability phenomenon is of the "density wave" type. Perturbations in the void fraction (density) propagate through the core and other two-phase regions at the vapor velocity, introducing phase lags in the pressure drops. These perturbations in void fraction are associated with corresponding changes in the neutron flux. Changes in the neutron flux are fed back to the fluid in the core region as changes in the heat flux. The heat flux perturbations are attenuated in magnitude by the thermal inertia of the fuel rods. They also lag the flux and void perturbations by an amount dependent on the time constant of the fuel rods for heat transfer. Three potential instability modes are considered in NEDC-32391P Table 2.3-5. Channel stability refers to the hydrodynamic stability of the fuel channel with the highest decay ratio. Channel stability is analyzed with no neutronics feedback and with a constant heat flux. It is not possible to produce

channel instability in a BWR core because of the neutronic coupling with adjacent channels. Channel stability can be a factor in exciting regional oscillations. Core wide stability refers to the excitation of the core in its fundamental neutronic mode. This is the most common type of instability observed in tests and a handful of events at operating BWRs. The perturbations in the flows and flux are in phase across the core. Core wide instability involves loop type perturbations, and the downcomer region participates in the process. In a regional instability, a higher order mode of the neutronics, with its associated subcriticality, is excited by the hydraulics. They are more likely to occur in a large core, which has a smaller subcriticality for the higher harmonics of the neutronics. The core pressure drop is essentially constant. The flows and fluxes are out of phase in different regions of the core. Regional instabilities have been observed in a few operating BWR cores. Both core wide and regional instabilities have been calculated with TRACG [NEDE-32177P]. In addition to channel and core stability, the overall plant stability is also analyzed. This examines the response of the plant to changes in set points of the control systems. Overall plant stability (generally an exercise in tuning the control systems) is not treated in this section.

A discussion of the individual items in NEDC-32391P Table 2.3-5 can be found in Supplement 1.

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

3.1 Introduction

This section describes the Bottom-Up process, one of two methods used to develop the test and analysis needs for SBWR. It complements the Top-Down process described in Chapter 2, with which it will be merged in Section 4.

The Top-Down process relies on the Phenomena Identification and Ranking Tables (PIRT) to identify thermal-hydraulic phenomena that are important for TRACG to model accurately and for which code qualification is required. The PIRT tables are an elaboration of the code qualification requirements from a microscopic or phenomenological point of view.

The Bottom-up approach compiles a list of SBWR-unique features, associated thermal-hydraulic phenomena and supporting TRACG qualification data from a macroscopic or system/component perspective. The purpose is to evaluate the adequacy of the data base used to qualify TRACG in the areas important to SBWR system thermal-hydraulic response.

3.2 Methodology

Each of the 127 SBWR systems was reviewed to determine if the system was unique, had unique features, or if a standard BWR component or system was subject to an application different from that found in the BWR operating fleet.

Those systems that did not directly affect the thermal-hydraulic response of the SBWR were not considered. System-unique features, the safety classification of the system, and the Master Parts List (MPL) number were documented. The principal design engineers were consulted with respect to the current reference system design and unique features, as well as References 3, 31, 32, to determine any new issues associated with that unique feature. For each of the issues, associated important thermal-hydraulic phenomena were identified.

The SBWR Product Structure is Shown in Figure 3.2-1.

3.3 Results

This section provides a summary of results.

3.3.1 RPV and Internals (B11)

Thirteen 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)

Nine 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)

Nine 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, 34 important containment system thermal-hydraulic phenomena were identified.

3.3.9 Passive Containment Cooling System (T15)

The systematic review of the SBWR design identified 17 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 High and Medium ranked 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 complete list and the correspondence between the PIRT and Bottom-Up items can be found in NEDC-32391P Table S1-9 of the supplement to this report. The phenomena that were ranked Medium have also been tabulated and have been tracked separately. The list is composed of separate tables for phenomena and interactions for each type of transient (LOCA, operational transients, ATWS, stability). 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 NEDC-32391P Section 5, these tables are compared against the Test Plan to confirm that all elements of the tables are covered by tests. Specific test coverage for the medium ranked phenomena is also addressed in NEDC-32391P Section 5. Where there is a lack of data, analysis will be performed.

4.1 Composite List of Identified Phenomena and Interactions

4.1.1 Loss-Of-Coolant Accident (LOCA)

This section discusses the phenomena and interactions important to LOCA.

4.1.2 Anticipated Transients

This section discusses the phenomena and interactions important to anticipated transients.

4.1.3 ATWS

PIRT for ATWS was developed in Subsections 2.3.3.

4.1.4 Stability

PIRT for stability was developed in Subsection 2.3.4.

4.2 Analytical Evaluation of System Interactions

The purpose of the system interaction study was: (1) to investigate the effects of both active and passive systems which could be available to support Engineered Systems Feature (ESF) systems during a LOCA; and, (2) to determine if interactions between the systems could degrade the performance of the ESF systems from what it would be if they were acting alone. The study extends earlier work presented in Chapter 6 of the SSAR (Reference 3), which evaluated the

effect of the break location and of various single failures. A part of this earlier study examined the possible adverse effect of reverse flow through the Isolation Condenser during an inadvertent opening of a DPV. Additional analysis in Chapter 19 of the SSAR (Reference 3) examined use of non-safety grade engineered systems to prevent core damage.

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

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

4.2.1 Accident Scenario Definition

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

4.2.2 Results from the Primary Systems Interactions Study

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

4.2.3 Results from the Containment Systems Interactions Study

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

4.2.4 Summary of System Interaction Studies

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

4.3 Summary of Evaluations

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

4.3.1 LOCA

Table 4.1-1a summarized the highly ranked phenomena for LOCA/ECCS. This table will be used in its entirety as a list of needs for qualification tests. Items listed as being moderately important in Table 4.1-1b will also be evaluated.

4.3.2 Transients

All issues but one will be carried forward to Section 5 as needs for TRACG qualification.

4.3.3 ATWS and Stability

For ATWS, the phenomena shown in Table 4.1-4a will be carried through to Section 5 for evaluation. Phenomena of moderate importance from Table 4.1-4b will also be evaluated. The majority of the phenomena are captured either by the Transient PIRT (neutronic and thermal hydraulic issues, Isolation Condenser, etc.) for the reactor parameters or by the Containment PIRT for SRV discharge to the suppression pool (critical flow, pool stratification and heatup, etc.). Natural circulation at low downcomer water levels leading to reverse bypass flow providing an internal natural circulation path inside the core shroud is covered under LOCA.

5.0 MATRIX OF TESTS NEEDED FOR SBWR PERFORMANCE ANALYSIS

The tables of highly ranked and medium ranked 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 the existing tests which could be used to qualify TRACG. The list of test data has been restricted to include only tests that have been analyzed, or will be analyzed, with the configuration-controlled version of TRACG. This results in elimination of data from a number of facilities (simulations of earlier BWR types) that have not been reanalyzed with the current version of the code. The early GIRAFFE data (Phase 1) are now being used for confirmatory purposes and have been marked as such. In a few cases, additional testing or qualification was proposed and incorporated in the Test Plan. The resulting matrix of tests needed for TRACG qualification is presented in this section. The tests have been divided into (1) Separate Effects Tests, (2) Component Performance Tests, (3) Integral System Tests, and (4) Operating Plant Data. The first two types of tests are suitable for model development, the latter two for checking the overall performance of the code.

Separate tables have been included for the High and Medium ranked phenomena. The Medium ranked phenomena are less important. Where test data are not available, analyses will be performed to evaluate their significance.

A separate PIRT was developed for stability in Section 2 and the High and Medium parameters identified in NEDC-32391P Section 4. However, in evaluating test coverage, these have been consolidated into three entries: ST1 for hydrodynamic or channel stability; ST2 for core wide stability; and ST3 for regional stability.

In the interest of conciseness, the tables in NEDC-32391P Section 5 do not include identification of the phases of the LOCA or specific transient for which data are needed. This information is contained in the previous sections. Also, the tables in NEDC-32391P Section 5 do not contain sufficient detail to verify that each referenced test provides data over the applicable range of SBWR conditions. For the major SBWR facilities such as GIST, PANDA, GIRAFFE, and PANTHERS, relevant information is in Appendix A. For other test facilities, details on range of test data will be provided in the TRACG Qualification Report.

5.1 Separate Effects Tests

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

5.2 Component Performance Tests

The distinction between component tests and separate effects tests is that the component tests focus on overall component performance. The level of instrumentation may not be as extensive as in a separate effects test. The distinction is often blurred. Both types of data have controlled boundary conditions and are suitable for model development as well as validation.

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 data-base.

Of special note are the DPV tests, where the blowdown capacities of full-scale DPVs were tested. Full-scale tests of an IC module are planned at the PANTHERS facility. A scaled model will be tested at PANDA.

On the containment side, two full-scale PCCS modules will be tested at PANTHERS. A 1/25 scale module will be tested in PANDA. A smaller module with three tubes has been tested in the GIRAFFE test facility by TOSHIBA (supporting data). A large amount of data exists on the early blowdown response of pressure suppression type containments.

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.

This section discusses the integral systems testing of the SBWR.

5.4 Plant Operating Data

The transient response of the SBWR is similar to that of other BWRs for operational transients in many respects. 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

NEDC-32391P Sections 5.1 through 5.4 identified 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 6.1-1 shows the "Road-Map" of how the new and existing test data support SBWR certification.

6.1 TRACG Qualification Plan

This section discusses work required to qualify TRACG for SBWR application.

The Analysis Plan in Appendix A identifies the specific tests for which blind predictions and post-test analysis will be performed.

6.2 Use of Data for TRACG Model Improvement and Validation

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

If changes are necessary to the TRACG models, a new version of the code will be created and brought to a controlled Level 2 status under the GE-NE quality assurance procedures. If changes in the nodalization are indicated, calculations affected by the changes will be redone and reverified.

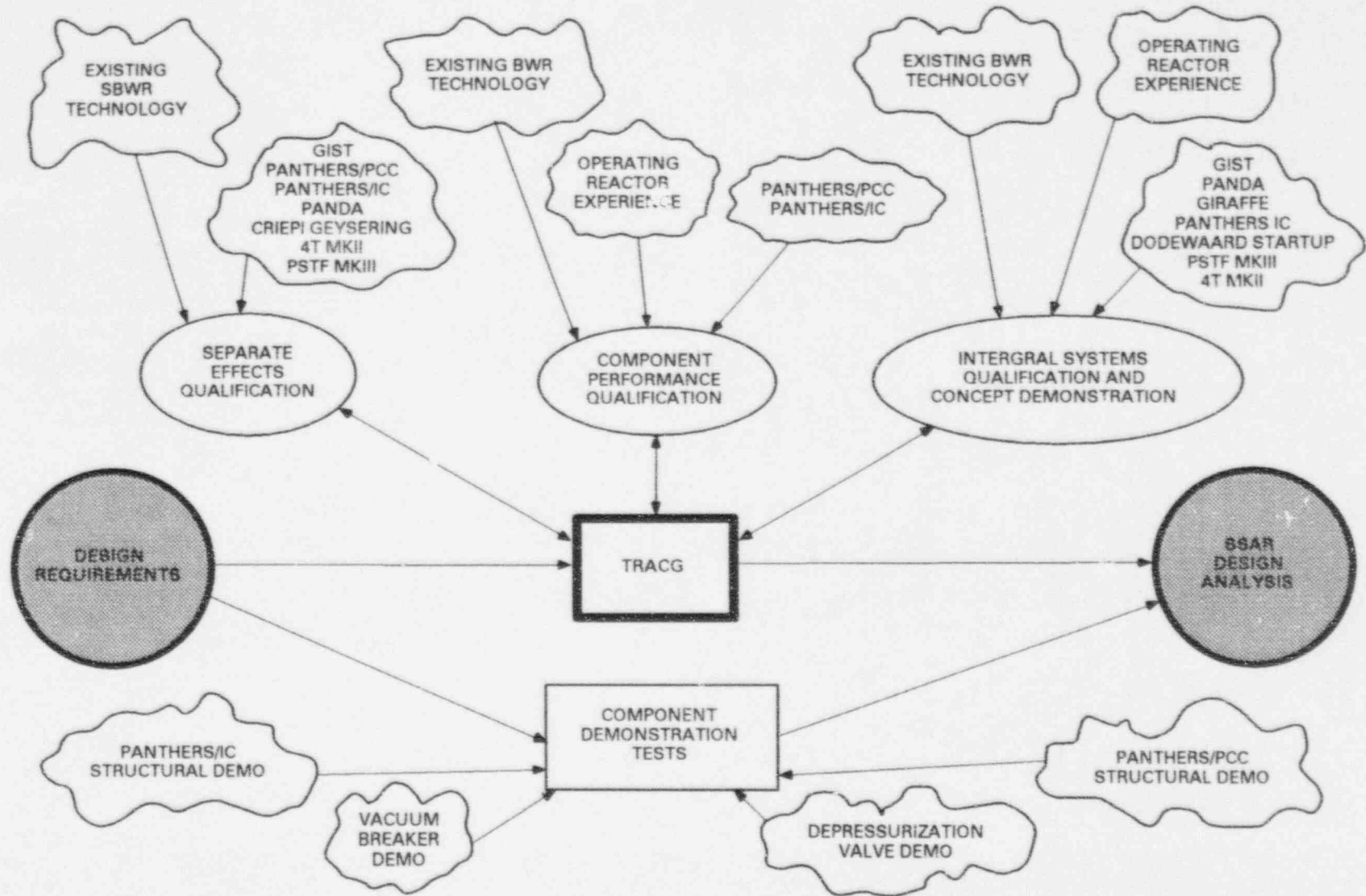


Figure 6.1-1 Technology Basis for SBWR Design

7.0 SUMMARY AND CONCLUSIONS

The Test and Analysis Program Description (TAPD) systematically defined test and analysis needs using Top-Down and Bottom-Up approaches to identify key phenomena, issues and interactions between phenomena and systems (Sections 2, 3, and 4 and Appendix C). These needs were compared to the existing test plan and the existing TRACG qualification plan, and modifications were made where necessary to fill in gaps in the database and the TRACG qualification base (Sections 5 and 6). The Test and Analysis Plan defined the remaining activities for closure (Appendix A). Test facility scaling was addressed quantitatively in Reference [32]. This document supersedes previous GE-NE submittals with regard to test objectives, test conditions, data use, and anticipated test analysis.

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

Test Plan

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

Analysis Plan

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

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

- 10CFR52.47 requires that "The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof." The studies summarized in Sections 2, 3 and 4 defined the phenomena important to SBWR safety in two independent ways. These are merged in Section 5 where the testing and experience bases applicable to each are shown. Each important phenomenon is covered by at least one separate effects test, component test, integral systems test, or operating reactor datum.
- 10 CFR52.47 requires that "Interdependent effects among the safety features of the design have been found to be acceptable by analysis, appropriate test programs, experience, or a combination thereof." The studies summarized in NEDC-32391P 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 NEDC-32391P Section 5 identifies elements which have been used to date (X entries), elements in which existing test data will be used (Q entries), and elements in which forthcoming test data will be used (T,Q entries) to qualify the SBWR analytical model, TRACG. These are collected in Section 6 to show the composite TRACG qualification plan.

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

8.0 REFERENCES

- [1] *TRACG Model Description*, J.G.M. Andersen, Md. Alamgir, Y.K. Cheung, L.A. Klebanov, J.C. Shaug. NEDE-32176P, Licensing Topical Report, January 1993.
- [2] *TRACG Qualification*, J.G.M. Andersen, M.D. Alamgir, J.S. Bowman, Y.K. Cheung, L.A. Klebanov, W. Marquino, M. Robergeau, D.A. Salmon, J.C. Shaug, B.S. Shiralkar, F.D. Shum, K.M. Vierow, NEDE-32177P, Licensing Topical Report, January 1993.
- [3] *SBWR Standard Safety Analysis Report*, 25A5113, Rev. A.
- [4] *Quantifying Reactor Safety Margins*, NUREG/CR/5249, EGG-2552, Rev. 4.
- [5] *Implementation of TRACG for Licensing Analysis*, J.L. Rash to R.C. Jones Jr. (NRC, NRR Chief Reactor Systems Branch), MFN 042-92 and JSC-92-010, March 9, 1992.
- [6] *Quantifying Reactor Safety Margins*, B.E. Boyack et. al, Nuclear Engineering and Design (Parts 1-4), 119, Elsevier Science Publishers B.V. (North Holland), 1990.
- [7] *Application of TRACG Model to SBWR Licensing Safety Analysis*, NEDE-32178P, H.T. Kim, February 1993.
- [8] *AEOD Concerns Regarding The Power Oscillation Event at LaSalle 2 (BWR-5)*, USNRC, AEOD Special Report S803, 1988.
- [9] *Leibstadt Stability Tests During Startup Testing* (contained in Reference 5).
- [10] *Stability Investigations of Forsmark-1 BWR*, B. Anderson, et. al., International Workshop on BWR Stability, October 1990.
- [11] *Vermont Yankee Cycle 8 Stability and Recirculation Pump Trip Test Report*, NEDE-25445, August 1982.
- [12] *ODYSY01/ODYSY02 Qualification Report*, NEDE-30227, (includes Peach Bottom-2 Stability Test Data), July 1983.
- [13] *The Startup of the Dodewaard Natural Circulation BWR – Experiences*, W.H.M. Nissen et. al., N.V. GKN, Netherlands, Waalbandijk 112a, 6669 MG Dodewaard (ANP '92), Tokyo, Japan (Detail measurements are contained in GKN-report 92-017/FY/R), October 25-29, 1992.
- [14] *Stability Monitoring of a Natural-Circulation-Cooled Boiling Water Reactor*, T.H.J.J. van der Hagen, Thesis, Delft Univ. of Technology, The Netherlands, 1989.
- [15] *Measurements at Various Pressures at the Dodewaard Natural Circulation Boiling Water Reactor in Cycle 23*, T.H.J.J. van der Hagen, GKN-Report 93-023/FY/R, 1993.
- [16] *Hatch Unit 2 Two-pump Trip Test*, November 18, 1978 (data contained in Reference 5).
- [17] *Hydrodynamic and Heat Transfer Measurements on a Full-Scale Simulated 36 Rod BWR Fuel Element with Non-Uniform Axial and Radial Heat Flux Distribution*, O. Nylund, et. al., FRIGG-4, ASEA-ATOM, December 1970.
- [18] *ABWR Horizontal Vent Containment Tests*, NEDC-31393 - SS Test Series.
- [19] *MIT and UCB Separate Effects Tests for PCCS Tube Geometry, "Single Tube Condensation Test Program"*, NEDC-32301.

- [20] *Mark III Confirmatory Test Program, Phase 1 Large Scale Demonstration Station Tests*, NEDM-13377, October 1974.
- [21] *MSC/NASTRAN Manual Version 67* Macneal-Schwendler Corporation, Los Angeles, CA.
- [22] *ABWR Horizontal Vent Containment Tests*, NEDC-31027 – FS series.
- [23] *Steam Purge Clearing Tests*, NEDE-28853.
- [24] *1/3 Area Scaled – Single Vent Row Tests*, NEDE-21596P.
- [25] *1/9 Area Scaled – 3 Vent Row Tests*, NEDE-24720P.
- [26] *JSBWR Phase 2 Report*, EBWR and VK50 Tests.
- [27] *Thermo-Hydraulic instability of natural circulation BWRs (Explanation on instability mechanisms at startup by homogeneous and thermodynamic equilibrium model considering flashing effect.)* by Fumio Inada and Tomio Ohkawa, *Int. Conf. New Trends in Nuclear System Thermohydraulics*, pp. 187-193, 1994.
- [28] *BWR 5 - 1/6 scale Boron Mixing Tests*, NEDE-22267.
- [29] *ABWR - 1/6 scale Boron Mixing Tests*, NEDC-30326.
- [30] Document No. BN-TOP-3, Bechtel Power Corp, San Francisco, CA, August 1975.
- [31] *Design and Analysis Similarities Between the ABWR and SBWR*, NEDC-32231, December 1993.
- [32] *Scaling of the SBWR Related Tests*, NEDC-32288 Rev. 1, R.E. Gamble, A. Hunsbedt, F.J. Moody, M.E. Parker, G. Yadigaroglu.
- [33] *Sixteen-Rod Heat Flux Investigation, Steam-water at 600 to 1250 psia*, E. Janssen, GE.
- [34] *The Effects of Lengths and Pressure on the Critical Heat Flux for a Closely Spaced 19-Rod Bundle in Forced Convective Boiling*, Dept. of Chem Eng., Engineering Research Lab Heat Transfer Research Facility, Columbia University, NY.
- [35] *NRC Requests for Additional Information (RAIs) on the Simplified Boiling Water Reactor (SBWR) Design*, Letter MFN No. 078-94 from P.W. Marriott (GE) to Richard W. Borchardt (USNRC), March 31, 1994.
- [36] *Quantifying Reactor Safety Margins — Application of Code Scaling Applicability and Uncertainty Evaluation Methodology for a Large-Break Loss-of-Coolant Accident*, B. Boyack et al., NUREG/CR-5249, 1989.
- [37] (See Reference 7).
- [38] *Mark II Pressure Suppression Test Program Phase II and III Tests*, NEDE-13442P-01, May 1976. *Mark II Pressure Suppression Test Program Phase II and III Tests*, NEDO-13468, October 1976.
- [39] *Hierarchical, Two-Tiered Scaling Analysis, Appendix D to An Integrated Structure and Scaling Methodology for Severe Accident Technical Issue Resolution*, Nuclear Regulatory Commission Report, NUREG/CR-5809, EGG-2659, November 1991.
- [40] GE COMPASS data base, run dated July 1994.
- [41] GE Master File Number MFN 042-92, Letter J.L. Raf'n (GE) to R.C. Jones (NRC), *Implementation of TRACG for Licensing Analysis*, March 9, 1992.

- [42] GIST Final Test Report, *SBWR Program Gravity-Driven Cooling System Integrated Systems Test*, GEFR-00850, October 1989.
- [43] *GIRAFFE Passive Heat Removal Testing Program*, by K.M. Vierow, NEDC-32215P, June 1993.
- [44] *Test Report for SBWR Depressurization Valve Operational and Flow Rate Test*, WYLE Report #41152-0, September 1990.
- [45] *Startup of the Dodewaard Natural Circulation Boiling Water Reactor*, Hagen, T.H.J.J. van der, Karuza, J., Nissen, W.H.M., Stekelenburg, A.J.C., Wouters, J.A.A., GKN Report 92-017/FY/R, 1992.
- [46] *Full-Scale Mark III Tests, Test Series 5707*, NEDE-21853-P, August 1978.
- [47] *Mark III Confirmatory Test Program, 1/3 Scale Condensation and Stratification Phenomena, Test Series 5807*, NEDE-21596P, March 1977.
- [48] *A Comparison of the RELAP Simulation of Mark III Suppression Pool Thermal Stratification with Data from the Pressure Suppression Test Facility*, NEDE-21957P, September 1978.
- [49] *Transient Behavior of Natural Circulation for Boiling Two-Phase Flow (2nd Report: Mechanism of Geysering)*, M. Aritomi, et al., Transactions of First JSME/ASME Joint International Conference on Nuclear Engineering (ICONE-1), Tokyo, Japan, pp. 87-94, November 4-7, 1991.
- [50] *PANTHERS Test Plan & Procedure*, SIET document No. 0098 PP 91, Revision 1.
- [51] *Technical Specification For IC & PCC Instruments Installation*, SIET document No. 00157 ST 92, Revision 1, January 1, 1994.
- [52] *PANTHERS-PCC Test Facility Instrumentation, Data Acquisition, & Processing Specification*, SIET document No. 00095 RS 91, Revision 1, June 8, 1994.
- [53] *Isolation Condenser and Passive Containment Condenser Test Requirements*, GE Nuclear Energy, 22A6999.
- [54] *PANDA Test Specification*, GE Nuclear Energy, 22A5587.
- [55] *PANDA Steady State PCC Performance Tests, Test Plan and Procedures*, Paul Scherrer Institut, document No. ALPHA-410.
- [56] *PANDA Pre-Test Analysis*, NUCON Report 40315-NUC-94-7034, GE Nuclear Energy Letter MFN 119-94.
- [57] *GIRAFFE Test Specification*, GE Nuclear Energy, 25A5677.
- [58] *Guide for Quality Assurance of Nuclear Power Plants*, Electrotechnical Standard Survey Committee, Japan Electric Association, JEAG 4101-1990.
- [59] *PANTHERS Pre-Test Calculation* (contained in Reference 35).
- [60] *Thermally Induced Flow Instabilities in Two-Phase Mixtures*, Ishii, M., and Zuber, N., 4th International Heat Transfer Conference, Paris, Paper No. B5.11, 1970.
- [61] *Drift Flux Model for Large Diameter Pipe and New Correlation for Pool Void Fraction*, Kataoka, I. and Ishii, M., Intl. J. Heat Mass Transfer, vol. 30, 1927-1939, 1987.

- [62] *Scaling and Analysis of Mixing in Large Stratified Volumes*, Peterson, P.F., Intl. J. Heat Mass Transfer, vol. 37, 97-106, 1994.
- [63] *The Thermal Hydraulics of a Boiling Water Reactor*, Lahey, Jr., R.T., Moody, F.J., American Nuclear Society, 1977.
- [64] *Thermo-hydraulic Instability of Natural Circulation BWRs at Low Pressure Start-up: Experimental Estimation of Instability Region with Test Facility Considering Scaling Law*, Fumio INADA, Masahiro FURUYA and Akira YASUO, Central Research Institute of Electric Power Industry, CRIEPI, Hiroaki TABATA and Yuzuru YOSHIOKA, Japan Atomic Power Company, JAPC, H. T. Kim, GE Nuclear Energy. Paper presented at ICONE3, April 1995.
- [65] *TRACG Analyses of Flashing Instability During Start-up*, Andersen, J.G.M. and Klebanov, L.A., GE Nuclear Energy. Paper presented at ICONE3, April 1995.
- [66] *Thermal-Hydraulic Oscillations in a Low Pressure Two-Phase Natural Circulation Loop at Low Powers and High Inlet Subcooling*, S.B. Wang, J.Y. Wu, Chin Pan and W.K. Lin, National Tsing Hua University, 4th International Topical Meeting on Nuclear Thermal Hydraulics, Operations and Safety, April 6-8, 1994.
- [67] *GIRAFFE Heat Removal Performance Tests Test Plan and Procedures*, Toshiba document no. TOGE110-TO7, revision 2, May 31, 1995.
- [68] *BWR Void Fraction Correlation*, J.A. Findlay, G.E. Dix, NEDE-21933, August, 1978.
- [69] *Critical Power and Pressure Drop Test-Step II Fuel Design Standardization Program for BWR/2-5 Reload Fuel*, B. Matzner and D.A. Wilhelmson, NEDC-31499P, San Jose, CA, 1987.
- [70] *Separate Effect Tests at UCB for PCCS Tube Geometry*, NEDC-32310.
- [71] *Backflow Leakage from the Bypass Region for ECCS Calculations*, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K, NEDE-21156, January, 1976.
- [72] *Supplemental Information for Plant Modifications to Eliminate Significant In-Core Vibrations*, NEDE-21156, January, 1976.
- [73] *Vacuum Breaker Design Basis Accident Simulation Test Procedure*, PCNVBR00001 Is.: 1 Rev. 1 (BPD DIFESA E SPAZIO Report).
- [74] *Monticello Inplant SRV Discharge High Load Test*, NEDC 20997, Rev. A.
- [75] *Caorso SRV Discharge Tests, Phase 2*, NEDO-25118.
- [76] *Caorso SRV Discharge Tests, Phase 1*, NEDE-25100P.
- [77] *PANTHERS - PCC Data Analysis Report*, SIET Document No. 0039 RA95, Rev. 0, June 20, 1995.
- [78] *Development of a System for Catalytic Hydrogen Reduction in Severe Accidents in Nuclear Power Plants (Title translated from the German)*, M. Eigenbauer, and M. Seidler, NIS Ingenieurgesellschaft Technical Report No. 1141/1866/0, Sep. 17, 1991.
- [79] Letter from J.E. Quinn (GE) to T.E. Quay (NRC), *SBWR - Pre-Test Analysis for PANDA Test M3*, MFN 161-95, August 21, 1995.

- [80] *PANTHERS/IC Test Plan*, SIET document no. 00396RI95.
- [81] *PANTHERS/IC Test Procedures*, SIET document no. 00395PP05.

APPENDIX A — TEST AND ANALYSIS PLAN (TAP)

A.1 Introduction

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

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

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

Test Plan

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

Analysis Plan

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

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

A.2 Test and Analysis Philosophy

A.2.1 Test Types

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

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

A.2.2 Test Overview

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

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

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

A.2.3 Test Approach

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

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

A.2.4 Analytical Approach

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

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

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

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

A.2.5 Documentation of Tests and Analysis

A.2.5.1 Test Documentation

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

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

A.2.5.2 Analysis Documentation

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

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

A.3 Test and Analysis Plan

A.3.1 Thermal-Hydraulic Tests

A.3.1.1 PANTHERS/PCC

A.3.1.1.1 Test Description

Overview

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

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

Instrumentation

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

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

Test Method

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

Steady state tests using a steam/air mixture are performed as follows. The test loop and PCC condenser are first purged with steam to remove any residual air from the system and to heat the PCC pool to saturation. When the pool is boiling, the required steam flow rate is established, followed by establishment of the required air flow rate to the PCC. The desired PCC inlet pressure

is then established by adjusting the position of the vent tank flow control valve. When steady conditions have been established, data is taken for a period of approximately 15 minutes.

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

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

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

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

A.3.1.1.2 Test Objectives

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

1. Demonstrate that the prototype PCC heat exchanger is capable of meeting its design requirements for heat rejection. (*Component Performance*)
2. Provide a sufficient data base 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

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

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

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

- Test Conditions 9, 15, 18, and 23 (Test Group P3) will be used to compare heat rejection rates over a range of air flow rates to the saturated, steam-only condition determined from Test Condition 41 in the pure steam series. Holding steam flow constant at near rated conditions, these tests yield the effect of air on the condensation process.
- Test Conditions 2, 13, 16, 17, 19, 22, and 25 (Test Group P4) supplement Test Group P3, in that they define condensation performance at the extremes of the SBWR air/steam mixture ranges, and at several intermediate points. These tests will be used to quantify 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, and 24. (Test Group P6) are lower priority tests. They are run at only a single inlet pressure to supplement the previously identified tests by increasing the data density within the already established air/steam flow map.

Transient Test Conditions

Table A.3-2c shows the PANTHERS/PCC Noncondensable Buildup Test Matrix. Eight test conditions are specified as Test Group P7. In these tests, steam is supplied at a constant rate, and steady state conditions are established in a manner similar to that of the steady state performance tests. Air, helium, or air/helium mixtures are then injected into the steam supply, with the vent line closed. The transient degradation in heat transfer performance will be measured, as a function of the total 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 52 and 53 are similar to Test Conditions 50 and 51, but with the steam supply superheated. Test Conditions 75 and 76 repeat Test Conditions 50 and 51, but utilize helium as the noncondensable gas instead of air. Helium is lighter than steam, and will mix in a manner similar to hydrogen. The results of Tests 50 through 53 plus 75 and 76 can be compared to establish performance differences between lighter-than-steam and heavier-than-steam gases as they build up in the heat exchanger tubes. Test Conditions 77 and 78 can be used to evaluate the effect of an air and helium mixture concurrently flowing into the heat exchanger.

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

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

- Test Conditions 54, 55, and 56 establish the effect of water level in the PCC pool for a range of steam and air/steam supply rates to the PCC. Data from Test Conditions 54, 55, and 56 can be compared to Test Conditions 41, 15, and 25, respectively, to obtain the effect of lowered water level on condensation performance. Test Conditions 54 and 55 can be compared to establish the effect of air content on the rate of pool boiloff.
- Test Groups P1 through P5, P7 and P8 provide a data base for TRACG qualification and meet Test Objective 2.

A.3.1.1.4 Justification of Test Conditions

PCC Operation

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

During a LOCA in the SBWR, the PCCS and the GDCS form a loop to keep the core covered with water and remove the decay heat. Steam coming off the core, leaves the RPV through the DPVs, enters the drywell, and flows to the PCCS. Condensate flow from the PCCS heat exchangers goes to the GDCS pool in the drywell. The GDCS delivers the water to the RPV where the decay heat of the core converts it to steam and starts the loop again.

PCC Operational Modes

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

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

Reference LOCA Condition —

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

PCC Capacity Greater Than The Decay Heat —

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

PCC Capacity Less Than The Decay Heat —

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

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

1. Pressure Drop Driven Mode

- PCC capacity \leq core decay heat
- PCC flow is forced by the DW to WW pressure difference
- PCC flow is a rich mixture of both steam and noncondensable gas

2. Condensation Pressure Driven Mode

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

PCC Purge and Vent Process

A PCCS purge event can occur as a result of the system being called upon to remove decay heat after an extended period of inactivity or by an increase in the mass fraction of noncondensable gas in the region of the drywell from which the system draws its inlet mixture. If the system is starting up after a period of inactivity, the condensers will contain a mixture of steam and noncondensable gas in near thermal equilibrium with the surrounding pool. The partial pressure of the steam will be approximately saturation pressure at the pool temperature and the remainder of the mixture will be noncondensable gas. This mixture must be expelled from the condensers before heat removal can begin. As steam is added to the drywell by the RPV, the drywell pressure will rise until the PCCS vents are cleared and the initial steam/noncondensable inventory of the condensers is vented. The movement of the initial inventory out of the condensers will be accompanied by ingestion of a fresh steam/noncondensable mixture at the existing drywell conditions in the neighborhood of the PCCS inlets. Depending upon the fraction of noncondensable in the inlet mixture, and the decay power, the system may or may not be able to condense steam at the rate it is being added to the drywell by the RPV.

Consider, first, the case where the PCCS heat removal rate at the existing inlet conditions is less than decay power. The situation is the same whether the PCCS is starting up from a period of inactivity or, while operating, is confronted with an increased noncondensable fraction in the inlet mixture. The drywell pressure will rise, thereby increasing the flow rate through the condensers from the drywell to the wetwell. The rise in drywell pressure also slightly increases the condensation rate. Additionally, as steam is continuously added to the drywell by the RPV, and a steam/gas mixture is transported through the condensers, the mass fraction of noncondensable in the inlet mixture will start to decrease. At some point, the combination of increasing drywell pressure and decreasing noncondensable inlet mass fraction enables the PCCS heat removal rate to match decay power and the drywell pressure stops rising.

Next, consider the case where PCCS heat removal rate at the existing inlet conditions is greater than decay power. Again, the situation is the same whether the PCCS is starting up from a period of inactivity or, while operating, is confronted with a decreased noncondensable fraction in the inlet mixture. The drywell pressure will start to drop, allowing water to reenter and close the vents. Unless the inlet conditions are pure steam, the PCCS will then start to accumulate noncondensable gas. Gas can accumulate in the vent pipes above the water level, in the headers,

and in the condenser tubes. The combination of accumulating noncondensable and, to a lesser extent, decreasing drywell pressure results in a decreasing condensation rate. Eventually, the condensation rate will drop below decay power and the drywell pressure will start to rise, initiating a new purge cycle.

The presence of vacuum breakers in the SBWR leads to a potential interaction between PCCS purging and vacuum breaker operation. As discussed above, if the instantaneous PCCS heat removal exceeds decay power, the drywell pressure will decrease. When the difference between DW and WW pressure drops below the submergence head of the PCCS vents, water will enter the vents and noncondensable will start to accumulate in the PCCS. The drywell pressure will continue to decrease until the combination of the lower pressure and the noncondensable accumulation drops the PCCS heat removal rate below decay power. If the PCCS noncondensable inventory when the vents close is relatively small, and the mass fraction of noncondensable in the inlet mixture is also small, the drywell pressure can continue to fall until it drops below the wetwell pressure by a sufficient amount to allow the vacuum breakers to open. The noncondensable which flows back to the drywell via the vacuum breakers increases the mass fraction of noncondensable in the inlet mixture, degrades condenser performance and leads to a new purge cycle. Thus, it can be seen that, depending on the attendant circumstances, a PCCS purge event may or may not lead to a vacuum breaker opening. In discussing these two possibilities, GE has introduced the nomenclature "strong purge" to identify a purge event which leads to opening of a vacuum breaker and "weak purge" to identify one which does not.

PANTHERS/PCC Operation

PANTHERS/PCC matches the behavior of the SBWR unit. Steam for the tests comes from the neighboring power plant. Instead of a GDCS, the facility has a condensate tank which collects the condensate from the PCC and returns it to the power plant. The boundary conditions on the condensate line match that for SBWR. The water level in the condensate tank is held to the same water level as found in the GDCS pool of SBWR. The pressure in the tank is equal to the steam inlet pressure which is the same as the SBWR where the pressure above the GDCS pool is the drywell pressure.

The PCC vent configuration differs among the types of tests and is discussed below.

Steady State Tests

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

Figure A.3-4 compares the range of test conditions for PANTHERS/PCC with the air and steam flow conditions for the SBWR main steamline and GDCS line break scenarios after one hour into a LOCA. The triangles representing the two breaks are constructed as the intersection of a vertical line, bounding the maximum steam flow, and a line drawn from the origin with a slope sufficient to envelope the calculated steam and noncondensable flow rates. The triangles are not one-to-one maps or time histories but, rather, bounds of the steam/noncondensable gas inlet conditions throughout the calculated SBWR LOCA scenario.

The triangles can be used to explain the progression of inlet conditions as the transient proceeds. This progression starts at the origin. In the period immediately following one hour, subcooled GDCS water is absorbing the decay heat power and there is no flow to the PCCS. This period is represented by the region near the origin. When the RPV water again reaches saturation, flow to the PCCS resumes and, at first, follows an approximately linear steam/noncondensable gas flow trajectory corresponding to the noncondensable gas mass fraction in the region of the drywell which feeds the PCCS. This initiates the purging process which transports the drywell noncondensable gas to the wetwell via the PCC units. At some point during the purge, the concentration of noncondensable gas in the drywell starts to drop and the steam/noncondensable gas trajectory turns over. The steam flow continues to increase as the noncondensable gas concentration in the inlet mixture decreases. The end of the purging process is represented by the extreme lower right corner of the triangle. The steam flow has now increased to its maximum value (matching the decay power) and the noncondensable gas flow has dropped to essentially zero. From this point, the steam flow "walks" backward along its axis as the decay power slowly drops.

The difference in the steam/noncondensable gas envelopes for the GDCS line and main steamline break accident scenarios results from the behavior during the GDCS injection phase of the transient. For the main steamline break, a large fraction of the subcooled GDCS water is retained in the pools as the RPV two-phase water level rapidly recovers to the main steamline elevation and equilibrates with the water in the GDCS pools. There is no vacuum breaker activation and, accordingly, there is a small noncondensable gas fraction in the drywell at the initiation of PCCS flow. PCCS flow initiates about one hour from the instant of LOCA at a decay power close to its rated heat removal capacity and, as a result of the low drywell noncondensable gas inventory, it rises to match decay power relatively rapidly. For the GDCS break, the pools drain completely and RPV steaming does not resume until about 2.5 hours from the LOCA. During the GDCS injection period, there are multiple vacuum breaker activations leading to a relatively large noncondensable gas fraction in the drywell when PCCS flow initiates. The larger PCCS steam flow does not match decay power until about 3.5 hours from the LOCA. PCCS steam flow is significantly reduced from the main steamline break case. A bottom drain line break would behave similarly to the GDCS line break.

There is no precise relationship between Figures A.3-4 and A.3-3. Figure A.3-3 illustrates various flow conditions which can exist in the PCCS circuit between the drywell and the wetwell. The purpose of Figure A.3-4 is to show that SBWR conditions are covered by the PANTHERS test matrix. Curves 1 through 4 in Figure A.3-3 describe a sequence of conditions which can follow the initial ascension of the PCCS heat removal to match decay power. At the end of the purge, drywell pressure exceeds wetwell pressure by slightly more than the head required to clear the PCCS vents. If the purging process has left the condensers in a relatively noncondensable gas-free condition (a "strong" purge), the drywell pressure will then start to fall as the PCCS removes slightly more than the decay power. Eventually, as it moves through the sequence from curves 1 to 4, the drywell pressure will decrease to the point where the vacuum breakers open. This will reintroduce noncondensable gas to the drywell and drive the PCCS inlet conditions to the left and upwards in the triangular regions of Figure A.3-4. Calculations performed to date have given no indication that this would lead to a "penetration" of the triangle boundary defined by the initial purge.

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

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

Transient Tests

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

Pressure Drop Driven and Condensation Pressure Driven Modes

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

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

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

A.3.1.1.5 TRACG Analysis Plan

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

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

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

Analysis results will be compared with test data as defined in Table A.3-3. For the steady state saturated and superheated steam conditions, the assessment of adequacy will be made on the basis of *total heat rejection rate* and *PCC pressure drop*. For air/steam and helium/steam mixtures, the *degradation factor*, defined as the ratio of the heat rejection rate in the 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, 76, and 78 addresses the buildup of noncondensibles in the PCC tubes, and will be predicted on a transient basis. Test Condition 51 uses air, Test Condition 76 uses helium, and Test Condition 78 uses both helium and air.

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

A.3.1.2 PANTHERS/IC

A.3.1.2.1 Test Description

Overview

PANTHERS/IC (Isolation Condenser) testing is performed at Societa Informazioni Esperienze Termoidrauliche (SIET) in Piacenza, Italy. The tests are performed in the same facility used for the PANTHERS/PCC program, but using several pieces of different equipment, in order to better simulate the performance environment of the IC. For the IC testing, the facility consists

of a prototype IC module, a steam supply vessel which simulates the SBWR reactor vessel, a vent volume, and associated piping sufficient to establish IC thermal-hydraulic performance. Both thermal-hydraulic and component demonstration tests are performed during these tests. This section covers the thermal-hydraulic portion of the testing; component structural performance tests are covered in Subsection A.3.2.2.

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

Instrumentation

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

Like the PCC testing, all test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. References 51 and 53 contain information on the IC instrumentation, as well as the PCC instrumentation. Additional information may be found in the PANTHER/IC Test Plan (Reference 80) and Test Procedures (Reference 81).

Test Method

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

The steam vessel and IC heat exchanger will be purged of initial air in a manner similar to that done with the PCC heat exchanger. The IC pressure will be at the design pressure or a lower value, depending on whether the test is also being used as a structural demonstration cycle. Subsection A.3.2.2 describes the PANTHERS/IC structural demonstration cycles. The IC is placed in operation by opening the IC drain valve. Steam supply to the steam vessel is then regulated such that the vessel pressure stabilizes at the desired value. Data will be acquired for a period of approximately 15 minutes. At this point, the steam supply can be increased or decreased to gather data at a different operating pressure, or testing may be terminated. In all cases, flow into the IC will be natural circulation driven, as is the case for the SBWR.

Noncondensable gas effects tests begin similarly until the point where pressure is stabilized at the desired value. For this case, a mixture of nitrogen and helium is injected into the IC supply line at a very low flow rate. The ratio of nitrogen to helium in the injected flow will be 3.5:1, simulating the composition of radiolytic gases. Gas injection will continue until the IC inlet pressure increases to 7.653 MPag (1110 psig). The noncondensable flow rate is approximately 3 to 5 g/s. The lower IC vent is then opened, and the IC vented until the pressure returns to the initial operating pressure, or stabilizes at an intermediate value. If the pressure returns to its initial value, the test is terminated. If the inlet pressure stabilizes, the IC top vent will be opened, and the

performance monitored until venting is complete, and the inlet pressure returns to the initial value. The test is then terminated.

Water level tests also begin with the IC in stable operation at the desired initial inlet pressure. The IC pool water level is then reduced and the IC performance monitored. Water level will be reduced until the IC inlet pressure reaches 8.618 MPag (1250 psig). The pool water level will then be increased to normal and IC performance allowed to return to normal. The test is then terminated.

A.3.1.2.2 Test Objectives

The objectives of the PANTHERS/IC Test Program are:

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

A.3.1.2.3 Test Matrix and Data Analysis

Steady State Performance Tests

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

Transient Test Conditions

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

Tables A.3-5b through A.3-5d give the PANTHERS/IC Transient Demonstration Test Matrix. Five Test Conditions are specified. Test Condition 1 (Test Group I2) is a set of two duplicate tests designed to demonstrate the startup and operation of the IC in a situation comparable to a reactor isolation and trip (Table A.3-5b). This is a type 2 test as shown in Figure A.3-23. Test Conditions 12 and 13 (Test Group I3) will have an air/helium mixture injected slowly after the steam vessel pressure has been reduced to the value specified as "inlet pressure" in Table A.3-5c. The IC will be vented when the inlet pressure reaches 7.655 MPag (1110 psig) or when the pressure peaks, if at a lower value. Re-establishment of condensation following venting

will be recorded. Test Conditions 14 and 15 (Test Group I4) are repeats of Test Conditions 12 and 13, but with the water level in the IC pool allowed to drop, exposing the IC tubes (Table A.3-5d). Both the IC pool level rate and the IC heat rejection rate will be monitored as a function of time.

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

A.3.1.2.4 Justification of Test Conditions

Steady State Tests

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

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

Transient Tests

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

For the pool water level tests, water levels at least as low as mid-height of the condenser tubes is specified provided the design pressure isn't exceeded. Bounding calculations based on decay heat rejection indicate that no more than one-third of the tubes may be uncovered during the 72 hour post scram period. Consequently, the defined testing bounds the SBWR range of conditions.

The PANTHERS/IC tests are component, not system, tests. The purpose of the transient tests is to measure the change in performance of the IC with (a) a known quantity of non-condensable gas present or (b) a change in pool water level. Although the test facility is similar to the arrangement found in the SBWR with the steam supply and condensate return line connected

to a large pressure vessel, the transient tests do not exactly match the system performance an IC would experience in the SBWR. For example, the operation of the heat exchanger in PANTHERS differs from the conditions it will encounter in the plant; i.e., steam and non-condensable gases are "metered" into the test facility, while in the plant the conditions at the inlet of the heat exchanger depend on the conditions in the RPV, and are not independent variables. However, the venting of the test unit will closely match the performance of the plant unit, and demonstrate that the IC can vent the gases and resume condensation.

A.3.1.2.5 TRACG Analysis Plan

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

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

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

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 Conditions 15 demonstrates TRACG capability to predict the effect of pool level on the degradation of IC performance.

A.3.1.3 PANDA

A.3.1.3.1 Test Description

Overview

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

heaters to simulate decay heat and thermal capacitance of the vessel and internals. The facility is capable of simulating SBWR accident scenarios starting approximately one hour into the LOCA.

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

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

Instrumentation

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

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

Test Method

Steady state Tests to demonstrate PCC performance will be the first tests performed in the PANDA facility. For these tests, the facility will be configured as described above and as shown schematically in Figure A.3-14.

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

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

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

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

A.3.1.3.2 Test Objectives

The test objectives of the PANDA Test Program are:

1. Provide additional data to: (a) support the adequacy of TRACG to predict the quasi-steady heat rejection rate of a PCC heat exchanger, and (b) identify the effects of scale on PCC performance. (*Steady State Separate Effects*)
2. Provide a sufficient data base 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. As noted in the test method section, the facility will be configured to inject known flow rates of saturated steam and air directly to the PCCS heat exchanger. The condenser inlet pressure will be maintained at approximately 300 kPa for tests with air injection by controlling the wetwell pressure. For tests with pure steam flow the condenser pressure will be allowed to come to the steady state equilibrium value consistent with the specified steam flow rate. The steam and air flow to the heat exchanger will be controlled and measured. In addition, the condenser drain flow will be measured.

Table A.3-9a shows the PANDA Steady State PCC Performance test matrix. In this series of tests, six test conditions (S1 through S6) 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.3-9a identifies the test conditions in PANDA and the corresponding PANTHERS and GIRAFFE Test Conditions.

- Five Test Conditions (Test Conditions S1 through S5) are planned with various air flows and a constant steam flow of 0.195 kg/sec. In addition, one test will be performed with a pure steam flow equivalent to that expected to match the steam condensing capacity of the condenser (Test Condition S6).

- 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.3-9a to meet the requirements of Test Objective 1(b).
- Tests S7 through S9 have been deleted.
- Tests S10, S11 and S12 are at the same conditions as Tests S3, S5 and S6, respectively. These three tests are to evaluate the repeatability of these earlier tests.
- Test S13 is at the same conditions as Tests S6 and S12, except that the PCC pool level is at the bottom of the upper header to evaluate the effect of reduced heat transfer from the upper header. For all other tests the PCC pool level is at the normal level.

Transient Integral Systems Tests

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

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

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

- *Test M1* was deleted and replaced with Test M10.
- *Test M2* is a perturbation to Test M3 with all of the break flow steam directed into drywell DW2. DW2 has two PCC condensers. This test maximizes the steam content of DW2 and the air content of DW1. It is the most asymmetric condition that can be established in PANDA. Test M2 results will be compared with Test M3 results to quantify asymmetric effects on PCCS containment performance.
- *Test M3* is the base case test, as defined in the previous paragraph.
- *Test M4* is a repeat of Test M3 to demonstrate transient system response repeatability.
- *Test M5* provides data for PCCS startup conditions similar to what might be expected in the SBWR following operation of the drywell spray. Test M5 will be initiated at the same conditions as Test M3. After one hour from the start of the test (i.e., two hours from the instant of the LOCA in the SBWR), conditions simulating the activation of

the SBWR drywell spray will be established. The PANDA capability for spray flow to the drywell is less than the scaled maximum flow rate for the drywell-spray mode of the SBWR FAPCS. Preliminary TRACG calculations for the PANDA facility have indicated that the available spray flow will not, by itself, be sufficient to cause the desired vacuum breaker action. Current plans are to supplement energy removal by the spray with a reduction (possibly to zero) of the heater power. TRACG calculations have also shown that the combination of the spray and the reduced power will achieve the desired objective. The plan is to continue the spray/power reduction for one hour with the intended result being a substantial increase in the drywell noncondensable inventory. The spray will then be turned off and the heater power ramped back to the decay power level. This will set the stage for PCCS restart with a relatively large noncondensable fraction in the inlet mixture.

- *Test M6* is a perturbation to Test M3 with the IC operating in parallel with the three PCC condensers throughout the test period. This test will provide data showing the interaction between the PCC condensers and the IC, as well as the effect of the additional heat removal by the IC on containment and reactor system performance.
- Test M7 will examine PCCS startup under conditions where the RPV is producing steam and the drywell is blanketed with noncondensable gas. The initial conditions for the PANDA vessels are given in Table A.3-10b. In the context of the SBWR, this set of conditions can be viewed as the limiting result of a combination of RPV and DW heat removal mechanisms (PCCS, ICS, sprays, etc.) which have caused all the noncondensable gas, transported from the drywell to the wetwell during the blowdown, to be redistributed back to the drywell.
- *Test M8* is a perturbation to Test M3, but with drywell-to-wetwell bypass leakage. The bypass leakage area will be set at ten times the allowable SBWR value as scaled to PANDA. This test will provide the effect of bypass leakage on containment performance.
- The original objective of PANDA Test M9 was to obtain conditions simulating the transition from the GDCS injection phase to the long-term cooling phase of the post-LOCA transient. This remains the objective of record, but it is currently being balanced against the capabilities of the PANDA facility and the possible desirability of examining an alternative parameter variation which may be of interest following the performance of earlier tests in the PANDA matrix. Consequently, there is no further specification of Test M9 available at this time.
- *Test M10* will have test conditions defined later, utilizing the experience gained from the previous tests. These test will focus specifically on systems interactions.
- PANDA tests M2 through M10 provide a data base for TRACG qualification that meets Test Objective 2.
- PANDA tests M2 through M10 address long-term operation of the PCCS. Tests M5 through M7, M9 and M10 address systems interaction and PCCS restart issues. These tests meet the requirements of Test Objective 3.

A.3.1.3.4 Justification of Test Conditions

Steady State Tests

The conditions specified for PANDA Tests S1 through S6 were tabulated in Table A.3-9a. As noted in Subsection A.2.3, the SBWR test program philosophy is to test in multiple scales, wherever feasible. As noted in Table A.3-9a, every PANDA steady state test shares test conditions with a condition from PANTHERS/PCC (see Subsection A.3.1.1) and GIRAFFE (see Subsection A.3.1.5). The specific conditions chosen duplicate PANTHERS/PCC Test Group P3 conditions at a steam flow near the mid-range for SBWR LOCA conditions and for a range of air flow fractions bounding the SBWR range. Additionally, one pure-steam test was chosen at near the maximum for the SBWR range.

PANDA Tests S1-S6 are shown on the SBWR flow map in Figure A.3-15. This figure may be compared with Figure A.3-4 to see the similarity to the PANTHERS/PCC matrix.

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

A.3.1.3.4.1 Transient Integral Systems Tests

Choice of the Base Case

The integral system response tests are specialized with the goal of investigating the highly ranked phenomena identified as Qualification Needs in NEDC-32391P Table 6.1-1. Since the number of tests is limited, the choice of conditions must be made to address the potential for systems interactions as well as individual system operations. Additionally, specific phenomena (e.g., drywell depressurization due to spray initiation, and PCC restart following noncondensable re-entry to the drywell) need to be addressed. Consequently, most integral systems test programs tend to be performed by definition of a *Base Case Test*, around which perturbations are made to assess the effects of specific systems, systems interactions, and phenomena of interest. This testing philosophy was chosen for the PANDA program.

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

Determination of Base Case Initial Conditions

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

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

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

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

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

Other Tests

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

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

A.3.1.3.5 TRACG Analysis Plan

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

- **Pure Steam Condensation** — Analysis of Tests S1 and S6 demonstrate TRACG's 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 noncondensable mass fraction in the PANDA PCC configuration.
- **Drywell-Wetwell Noncondensable Distribution** — Analysis of Tests M2, M3, and M5 through M10 addresses the effects of initial gas and vapor distribution within the containment system, including vacuum breaker flow, and demonstrate TRACG's capability to model integral systems performance.
- **Systems Interactions** — Analytical studies of systems interactions have identified vacuum breaker and IC operation as the most likely candidates for systems interaction effects. Analysis of Tests M5 and M6 address TRACG's capability to model systems interactions.
- **Bypass Leakage** — The TRACG analysis of Test M8 provides qualification of bypass leakage modeling.

A3.1.4 GIST

A3.1.4.1 Facility Description

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

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

Key interconnecting lines, such as drywell vents and depressurization lines with quenchers, were also included. The suppression pool/wetwell includes the water supply tank, a recirculation

pump system used to heat and cool the pool water, and the air lines for pressurizing the wetwell air space.

The GIST facility was a simulation of the SBWR design as it existed in 1988. Several differences exist between the GIST configuration and the final SBWR design. These differences are listed and reconciled in Reference 32. These differences notwithstanding, the facility simulates all important GDCS refilling phenomena. The data on GDCS initiation, flow rates, and chimney and downcomer level response can be used for TRACG qualification. (Additional data are being obtained from the GIRAFFE/SIT tests, Subsection A.3.1.7).

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*) This test addresses the key interaction between XL3 and phenomena E1, E2, E3, E7 and F1 listed in NEDC-32391P Table 5.3-1a

A3.1.4.3 Test Matrix

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

- Bottom Drain Line Break (BDLB)
- Main 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 was performed and then subsequent tests were run where only one parameter at a time was varied from that used in the base case. The GIST facility modeled SBWR plant behavior during the final stages of the RPV blowdown. The tests started with the vessel at 100 psig and continued until the GDCS flow initiated and flooded the RPV.

- *Series BDLB* (Bottom Drain Line Break) consisted of parametric variations around the base test case of a relatively small break below the core. Seven tests were run in this configuration.
- *Series MSLB* (Main Steamline Break) consisted of eight tests, six of which were parametric variations and two of which were duplicates to establish the repeatability of results.

- *Series GDLB* (GDCS Line Break) consisted of four tests. Variations in ADS configuration were the parameter in this series.
- *Series NB* (No-Break) consisted of seven tests. This series typically utilized conditions well removed from the SBWR 1988 design envelope. They form a data set at or outside the limits of SBWR, and are the most challenging for TRACG analysis. For example, this series included several tests where the wetwell initial pressure was atmospheric, and no air-purge occurred since there was no break. The major difference between the 1988 GIST and current SBWR configurations is the location of the GDCS pool. From the standpoint of GDCS injection, the GIST configuration is conservative relative to the SBWR because the GDCS driving head is always slightly less in GIST than in the SBWR. In the case of zero wetwell pressure, the GDCS injection head is much less than in the SBWR. This makes GDCS injection in GIST more challenging.
- Analysis of GIST data as reported in Reference 42 has proven the technical feasibility of the GDCS concept and accomplishes Test Objective 1.
- The overall GIST data base provides a sufficient basis for TRACG qualification and accomplishes Test Objective 2.

A.3.1.4.4 TRACG Analysis Plan

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

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

A.3.1.5 GIRAFFE

A.3.1.5.1 Test Description

Overview

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

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

These GIRAFFE tests were performed as developmental tests. Except for comparison of the GIRAFFE steady state PCC performance data with that from PANTHERS and PANDA, no additional analysis of this data is planned.

A.3.1.5.2 Test Objectives

The objectives of the GIRAFFE Test Program are:

1. Provide a data base 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 data base to support primary data taken at other scales to confirm the capability of TRACG to predict PCCS system performance. (*Integral Systems Tests*)

A.3.1.5.3 Test Matrix and Data Analysis

Steady State Tests

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

Table A.3-16 shows the GIRAFFE PCC Steady State Performance Matrix used to provide data in support of the test objectives. Thirteen test conditions are included. These tests are identified in the test report as the Phase 1, Step 1 Tests, and comprise Phase 1 Test group. 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 Phase 1 Test group provide a support data base for TRACG qualification and meet the requirements of Test Objective 1. Data from Phase 1 Test group will be compared to that from corresponding PANDA and PANTHERS tests to corroborate those results at a third scale.

A3.1.5.4 TRACG Analysis

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

A.3.1.6 GIRAFFE/Helium

A.3.1.6.1 Test Description

Overview

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

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

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

Instrumentation

Instrumentation utilized in the GIRAFFE/Helium test program is similar to that used in earlier GIRAFFE tests. Test instrumentation consists of 81 thermocouple measurements, 5 pressure measurements, 19 differential pressure measurements, and 4 flow rate measurements. Test instrumentation is calibrated against standards equivalent to the U.S. National Institute of Standards and Technology. Detail of the instrumentation, including instrument lists, types, and ranges are included in the Test Plan and Procedure (Reference 67).

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

This data will be used to validate indirect measurements of noncondensable concentration inferred from temperature measurements.

Method

GIRAFFE/Helium testing follows a methodology very similar to that used in PANDA. Once the initial conditions for a given test have been established, all control (except for the decay of RPV power and helium injection, if called for) will be terminated, and the GIRAFFE containment will be allowed to function without operator intervention (except that the vacuum breaker is operated manually to simulate automatic operation in SBWR), mirroring the SSAR assumptions for the SBWR. Details are included in the Test Plan and Procedure (Reference 67) for these tests.

A.3.1.6.2 Test Objectives

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

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

A.3.1.6.3 Test Matrix and Data Analysis

Helium Test Series

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

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

- Test H1 is the base case with nominal initial conditions the same as in PANDA Test M3. Initial conditions are given in Table A.3-18
- Test H2 is a repeat of Test H1, but with helium replacing the total volume of nitrogen in the drywell and PCCS.
- Test H3 will have the same initial total drywell pressure as Tests H1 and H2, but with the initial noncondensable fraction consisting of a helium/nitrogen mixture.

- Test H4 will start with the same initial drywell conditions as Test H1, and will have constant helium injection to the drywell. The helium addition rate will be such that the helium is injected over a period of one hour. The helium injection will be terminated when the total mass of helium added is equal to the initial drywell helium mass for Test H3.

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

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

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

"Tie-Back" Test Series

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

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

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

A.3.1.6.4 Justification of Test Conditions

Choice of the Base Case

Test H1, defined as the Base Case for Test Groups H and T, utilizes the same initial conditions as PANDA Test M3 (see Table A.3-10a). The justification for the M3 conditions given in Subsection A.3.1.3.4 also apply to GIRAFFE/Helium test H1.

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

Other Tests

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

Test H2 helium specification is unrelated to any scenario; all the nitrogen used in Test H1 is replaced with helium to obtain a one-to-one comparison of PCC system performance in the presence of lighter-than-steam and heavier-than-steam noncondensibles. The purpose of the H3 and H4 tests is to demonstrate the effect of a high concentration of a lighter than steam gas on the performance of the PCC. At the same time, the pressure capability of the facility must be considered. The design pressure for the drywell and suppression chamber is 0.6 MPa. In order to assure that the design pressure is not exceeded during Tests H3 and H4, a helium mass less than the scaled amount determined for a 100% fuel-clad metal water reaction is used. Therefore, Tests H3 and H4 will utilize a helium mass of 0.97 kilograms, which is equivalent to 20% by volume of the scaled amount of hydrogen gas that would be generated by a 100% fuel-clad metal water reaction. This results in an initial GIRAFFE drywell concentration of approximately 23% helium by volume. Since this quantity is equal to approximately 60 times the PCC volume it is a sufficiently high quantity of helium to capture the prototypical behavior of lighter-than-steam gases on the performance of the PCC.

Test T1 initial conditions are the same as the initial conditions used for the previous GIRAFFE main steamline break test. For Test T2, the total nitrogen mass for the drywell and suppression chamber is equal to the total nitrogen mass for Test H1. The initial drywell nitrogen mass for Test T2 is approximately midway between that for Tests H1 and T1. Therefore, in Test T2 the effect of the nitrogen distribution between the drywell and suppression chamber will be investigated.

A.3.1.6.5 TRACG Analysis Plans

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

A.3.1.7 GIRAFFE/SIT (Systems Interaction Test)

A.3.1.7.1 Test Description

Overview

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

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

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

Instrumentation

Instrumentation utilized in the GIRAFFE/SIT test program is similar to that used in earlier GIRAFFE tests (see Subsection A.3.1.6.1). Added instrumentation for the IC unit includes tube, steam box and water box fluid temperatures; steam supply line flow; steam box pressure; and collapsed water levels in the IC pool, IC tube and water box. The collapsed chimney water levels are measured over 4 intervals using pressure taps at 0.5 to 7.0m above the top of the heater section. Both the bypass region collapsed level and the downcomer collapsed level are measured. Instrumentation details, including instrument lists, types, and range will be included in the test plan and procedures for these tests.

Method

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

A.3.1.7.2 Test Objectives

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

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

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

A.3.1.7.3 Test Matrix and Data Analysis

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

The initial conditions for all tests approximate SBWR conditions approximately 10 minutes post-LOCA, based on the breaks and equipment operations listed in Table A.3-21. All tests will run for approximately two hours. Containment related parameters will be based on the appropriate SBWR TRACG LOCA case at the time RPV pressure is 1.034 MPa (150 psia). Heater power will be decayed from this time to simulate decay heat and stored energy transfer.

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

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

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

- Test GS1 is the base case test, a GDCS line break, with DPV failure as the single failure and neither the PCCS, nor the IC, in operation. This test has initial conditions similar to GIST Test C01A, and may be compared with GIST C01A to evaluate the effects of configuration distortions in GIST and potential GDCS containment system performance interactions.
- Test GS2 is the same as Test G1, except that the PCCS and IC are operating. Test GS2 results will be compared to those of Test GS1 for identification of potential systems interactions associated with the IC and PCCS.
- Test GS3 is a bottom drain line break with DPV failure. For this test both the PCCS and IC will be functioning. Data from Test GS3 will be examined for identification of potential systems interactions associated with the IC and PCC for a bottom drain line break.
- Test GS4 is a GDCS line break, with the single failure being a GDCS valve failure in one of the other GDCS injection lines. As in Test GS3, both the PCC and IC will be in operation. This condition is expected to provide the slowest rate of recovery for the chimney swollen water level. Data from test GS4 can be compared to test GS1 to identify potential interactions with the IC and PCC even though the single failures are different.

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

A.3.1.7.4 Justification of Test Conditions

Choice of the Base Case Test

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

Other Tests

The other test cases were defined with the objective of identifying systems interactions, should they occur. Since the primary focus of this testing is GDCS performance, the RPV chimney swollen water level is the figure of merit in these investigations. SBWR TRACG predictions for several break locations, single failures, and IC/PCC operation combinations were performed. The additional tests, which are presented in Table A.3-23, were chosen based on the TRACG results. The "objective" column in this table indicates the major feature of each case which will be captured by the test. The rate of chimney swollen water level recovery after the time of minimum level is expected to be substantially different for the GDL and BDL break cases noted for those "recovery" objectives. Some SBWR TRACG cases evidenced chimney swollen water level oscillations which is attributed to GDCS flow quenching voids in the RPV and to break flow depressurizing the drywell. Thus, a wide range of conditions is represented by the four test cases selected.

Test Initiation

The tests will be initiated from a steady state condition at the time RPV pressure reaches 1.034 MPa. This is unlike the SBWR situation, where the accident starts at much higher pressures. This difference is justified since the void formation, which results from depressurization, occurs very soon after the pressure is reduced to the fluid saturation temperature. SBWR calculations indicate that void fractions increase in about 20 seconds to much larger values, then remain relatively constant until GDCS flow is injected on the order of 80 seconds later (causing some void collapse) or until pressures become much lower than 1.034 MPa. Preliminary GIRAFFE TRACG cases indicate a very similar behavior. Based on this general agreement, the selected strategy is appropriate. The water level instrumentation in the test facility can be used to determine variations in void fraction, and this will be used to provide test-related confirmation of the strategy.

A.3.1.7.5 TRACG Analysis Plan

All four transient tests in the GIRAFFE/SIT series will have TRACG analysis performed on a blind post test basis. Although the tests will be performed prior to the TRACG analysis, the analyst will have no knowledge of the test results while the analysis is being performed. Exceptions will be information needed to conduct the analysis such as actual initial conditions,

heater power and microheater power during the test. The assessment of TRACG's adequacy will be based on the ability to predict chimney and downcomer swollen and collapsed water level.

A.3.1.8 Other Analyses Planned

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

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

A.3.1.8.1 1/6 Scale Boron Mixing Test

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

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

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

A.3.1.8.2 CRIEPI Natural Circulation Thermal-Hydraulic Test Facility

The CRIEPI 7 MPa test facility is a parallel channel test facility intended to study the stability characteristics of a natural circulation loop during startup conditions. Figure A.3-24 shows the configuration of the test facility. 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. Flow measurements (natural circulation conditions) for different system pressure, power and channel inlet temperatures were obtained at the CRIEPI experimental facility. Instability similar to geysering was observed at some conditions.

Test Conditions and Scalability to SBWR

The results of the CRIEPI test and comparisons with TRACG are reported in several papers (References 64, 27, and 65).

The basic equation of the drift-flux model were non-dimensionalized to arrive at the important non-dimensional numbers for the hydrodynamic stability. The characteristic numbers are reported in Table A.3-29 for the SBWR and CRIEPI facility. The tests were not run at the full power conditions shown in the table. The full power conditions were selected to match N_{pch} and N_{sub} of the SBWR at full power. To arrive at the low power conditions, the power of the facility was ratioed down by the same amount as the SBWR for the desired conditions. The subcooling was set to match the SBWR also. A complete discussion of the method is described in Reference 64.

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

Tests were run at pressures of 0.1, 0.2, 0.35 and 0.5 MPa. The results are shown in Figure A.3-25. The figure shows the instability region in the heat flux-channel inlet subcooling plane as developed in Reference 64. Additionally, the expected SBWR conditions during start-up for these pressures are shown on the figure. The results indicate a significant amount of margin to unstable behavior in the SBWR. The margin increases as the pressure is increased.

Some additional tests in another facility showed unstable behavior over the entire range of conditions tested there (Reference 66). However, these tests were run at a much higher heat flux and subcooling than is representative of the SBWR. Figure A.3-25 shows that instability at higher heat flux and subcooling is consistent with the unstable region for the CRIEPI results.

A.3.1.8.3 Dodewaard Plant Startup

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

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

TRACG analysis of this startup is being performed.

A.3.1.8.4 Containment System Response — PSTF Mark III

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

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

A.3.1.8.5 Containment System Response — Mark II 4T

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

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

These three tests will be analyzed with TRACG.

A.3.1.8.6 Suppression Pool Stratification — PSTF

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

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

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

A.3.2 Component Demonstration Testing

A.3.2.1 PANTHERS/PCC

A.3.2.1.1 Test Description

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

A.3.2.1.2 Test Objectives

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

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

A.3.2.1.3 Test Matrix and Data Analysis

The approach taken to address the test objective is to subject the equipment to a total number of pressure and temperature cycles well in excess of that expected over the anticipated SBWR lifetime. The test matrix is shown in Table A.3-25. The number of cycles was conservatively chosen as 10 LOCA cycles and 300 pressure test cycles. This represents five times the design requirement number of hypothetical LOCAs (2) and nearly 17 times the number of expected pneumatic test cycles in accordance with 10CFR50, Appendix J over the 60-year design life of the PCC. Note, no credit is taken for the thermal cycles experienced during the PCC thermal-hydraulic testing in determination of this Component Demonstration Test Matrix.

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

Simulated LOCA Pressurizations

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

Simulated Pneumatic Leak Test Pressurizations

Simulated pneumatic tests are performed by pressurizing the PCC heat exchanger with air to 758 kPag (110 psig). The PCC pool temperature is at ambient conditions during these pressurizations. The test pressure is held for 2 minutes for each cycle. A total of 300 cycles are

performed. The test data will be analyzed by review of strains and acceleration data against component acceptance requirements, both in terms of magnitude and frequency content.

A.3.2.2 PANTHERS/IC

A.3.2.2.1 Test Description

Overview

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

Test Method

Figure A.3-23 illustrates the four cycle types for the PANTHERS/IC Component Demonstration tests. For each cycle, the IC unit is initially brought to high pressure by purging the air with high pressure steam. The rate of pressure increase simulates the SBWR startup, and is limited to prevent undue thermal stresses. The drain valve is closed, and the unit fills with water to the level of the IC pool. The IC pool is at ambient conditions. This simulates the standby mode of the IC in a SBWR when the plant is in operation. The tests begin when the unit has stabilized at its required initial inlet pressure (P1).

In Types 2 and 5 tests, the IC drain valve is opened, allowing the water to flow from the unit. This exposes the tubes to steam and begins the IC operation. The operator controls the steam flow to the steam vessel to bring the unit to pressure P2 and holds it for two hours. Following this, the unit is cooled down at a controlled rate without condensing steam. Type 2 test is the same as the Transient Demonstration Test Group I2 (Subsection A.3.1.2.3), which demonstrates the startup and operation of the IC unit. Type 5 tests are similar to Type 2 tests, but use a different initial pressure.

The Type 6 cycle simulates the pressures and temperatures the IC would experience during normal plant startup and shutdown. From the initial inlet pressure of P1, the unit is cooled down at a controlled rate without condensing steam.

The Type 7 cycle simulates an ATWS event. From pressure P1, the drain valve is opened and the IC begins operation. The operator briefly brings the unit to a high pressure (P2), which is similar to what an IC unit would experience at the start of an ATWS event. The pressure is then brought back to initial pressure or as close as is possible (P3) and held for two hours. Following this, the unit is cooled down at a controlled rate.

A.3.2.2.2 Test Objectives

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

Confirm that the mechanical design of the IC heat exchanger is adequate by assuring that the loads used in design envelope the loads expected during the SBWR service conditions.

A.3.2.2.3 Test Matrix and Data Analysis

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

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

The test measurements will then be used in conjunction with other loads in an analysis to calculate the resultant stresses and cumulative fatigue effects to ensuring conformance with the ASME code.

A.3.2.3 Depressurization Valve (DPV)

A.3.2.3.1 Test Description

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

A.3.2.3.2 Test Objectives

The test objectives of the DPV Test Program were:

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

A.3.2.3.3 Test Matrix and Data Analysis

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

Two full-scale prototype squib actuated DPVs were manufactured, assembled and tested by Pyronetics Devices, Inc., a subsidiary of OEA, Inc., of Denver Colorado. Firing tests were performed on a full-scale valve under both a high pressure (1500 psig) condition at the valve inlet and a low pressure (1 psig) condition at the valve inlet. A momentary electrical signal was supplied and it was confirmed that the valve opened within the required response time and remained open without an external power source. A thermal exposure heat transfer test was performed on the valve to assess the effects of ambient temperature and 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 steamline section. The DPV was connected to this prototypical section and instrumented with pressure, temperature, and strain gages, accelerometers and displacement transducers. Four steam blowdown tests were performed. The test data confirmed that the DPV mass flow rate would be on the order of 2.4×10^6 lbm/hr at an operating pressure of 1100 psia.

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

A.3.2.4 Vacuum Breaker Valve

A.3.2.4.1 Test Description

The vacuum breaker valve test program was designed to confirm that the vacuum breaker valve would provide a reliable leak tight boundary between the drywell and wetwell and prevent the pressure in the wetwell from exceeding that of the drywell by more than three pounds per square inch. Leak tightness is achieved by use of a nonmetallic main seal and a backup hard seat. The double seal design provides assurance that maximum leakage requirements will not be exceeded in the event that an obstruction should lodge on either seat. A full scale prototype valve

was built and subjected to flow testing to verify lift pressure, flow capacity, and stability at low flow. The primary nonmetallic seal was radiation and thermally aged. Following thermal aging, the valve was dynamically aged and subjected to design basis accident conditions to confirm its leak tightness to steam. Finally, the fully aged valve was subjected to reliability testing to confirm that its intrinsic reliability was consistent with the assumptions of the SBWR PRA.

A.3.2.4.2 Test Objectives

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

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

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 lift pressure and flow stability met requirements. The flow test demonstrated that the valve stroke was not sufficient to meet minimum flow requirements. Since the natural stability of the 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 results in achieving the required flow performance. A seal was then aged with radiation and placed in the valve for thermal aging. The valve leak test was then repeated and it was shown that the seal was air bubble tight.

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

The Design Basis Accident test demonstrated that the fully aged valve meets leak requirements at steam pressures and temperatures characteristic of a loss-of-coolant accident followed by water spray. The leak tightness of the valve was demonstrated by measuring the

condensate from the steam that passed through the valve seals. During pressure peaks, water sprays and 80 hours of endurance testing, no measurable condensate leaked through the valve. The test demonstrated the inherent steam leak resistance of the valve.

The final test was the reliability testing, which subjected the fully-aged valve to grit ingestion to simulate possible environmental conditions that could affect bearing surfaces and seals during normal service. The valve was cycled three thousand times to demonstrate reliability at its required statistical failure rate of 3×10^{-4} per demand.

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

Facility	Data Group	Test Conditions	Description
PANTHERS/PCC	P1	7	PCC steady state performance; saturated steam
PANTHERS/PCC	P2	6	PCC steady state performance; superheated steam
PANTHERS/PCC	P3	4**	PCC steady state performance; air/steam mixtures
PANTHERS/PCC	P4	7**	PCC steady state performance; air/steam mixtures
PANTHERS/PCC	P5	2**	PCC steady state performance; air/steam mixtures
PANTHERS/PCC	P6	14	PCC steady state performance; air/steam mixtures
PANTHERS/PCC	P7	6	PCC performance; 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 start-up demonstration
PANTHERS/IC	I3	2	IC restart demonstration, noncondensable venting
PANTHERS/IC	I4	2	IC performance; water level effects
PANDA/PCC	S	7	PCC steady state performance; steam and air/steam mixtures
PANDA	M3, 4, 7	3	Containment performance
PANDA	M5, 6, 8	3	Containment performance
PANDA	M2, 9, 10	3	Containment performance
GIRAFFE	Phase 1	13	PCC steady state performance - steam and air/steam mixtures
GIRAFFE/Helium	H	4	Containment performance - noncondensable density effects
GIRAFFE/Helium	T	2	Containment performance - "Tie-back" test
GIRAFFE/SIT	GS	4	GDCS performance - integral systems tests
GIST	BDLB	7	GDCS performance - integrated system effects - bottom drain
GIST	MSLB	8	GDCS performance - integrated system effects - main steam
GIST	GDLB	4	GDCS performance - integrated system effects - GDCS breaks
GIST	NB	7	GDCS performance - integrated system effects - transients

* Test to be performed twice to demonstrate repeatability.

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

Table A.2-2 SBWR Test Documentation Submittals

Test	Submittal Title	Document No.	Actual Submittal Date
PANDA	Test Specification	22A5587 Rev. 1	15 Feb 95
	As-Built Drawing Package	MFN 044-95	27 Mar 95
	Instrumentation Drawing Package		
	QA Implementation Procedures	PPCP-QA-01	16 Feb 95
	Pre-Test Analysis (S1-S6)	40315-NUC-94-7034	27 Sept 94
	Test Plan and Procedures (S1-S9)	ALPHA-410-0	20 April 95
	Apparent Test Results (S1-S6)	ALPHA-509-0	20 July 95
	Data Transmittal Report (S1-S6)		
	Test Plan and Procedure (M3, 4, 7)		
	Pre-test Analysis M3	MFN 161-95	21 Aug 95
	Apparent Test Results (M3, 4, 7)		
	Data Transmittal Report (M3, 4, 7)		
	Test Plan and Procedure (M5, 6, 8)		
	Pre-Test Analysis M5		
	Apparent Test Results (M5, 6, 8)		
	Data Transmittal Report (M5, 6, 8)		
	Test Specification (update for M1&9)	22A5587 Rev. 2	
	Test Plan and Procedure (M1, 2, 9)		
	Pre-Test Analysis M2, M9		
	Apparent Test Results (M1, 2, 9)		
Data Transmittal Report (M1, 2, 9)			
PANDA Data Analysis Report			
GIRAFFE/Helium	Test Specification	25A5677 Rev. 1	17 July 95
	As-Built Drawing Package	TOGE110-T19 Rev. 0 (MFN 124-95)	24 July 95
	Instrumentation Drawing Package	TOGE110-T07 Rev. 2	14 July 95
	QA Plan	TOGE110-T01 Rev. 1	27 April 95
	Test Plan and Procedures (T1, H1-H4, T2)	TOGE110-T07 Rev. 2	14 July 95
	Apparent Test Results (H1, H2)		
	Pre-test Analysis (H1-H4)	MFN 159-95	18 Aug 95
	Data Transmittal Report (H1, H2)		
	Apparent Test Results (H3, H4)		
	Apparent Test Results (T1, T2)		
Data Transmittal Report (H3, H4, T1, T2)			

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

Test	Submittal Title	Document No.	Actual Submittal Date
GIRAFFE/SIT	Test Specification	MFN 144-95	8 Aug 95
	Test Plan and Procedures (GS1-GS4)		
	Apparent Test Results		
	Data Transmittal Report		
	GIRAFFE Data Analysis Report		
PANTHERS/PCC	Test Specification	23A6999 Rev.3	15 Feb 95
	As-Built Drawing Package	many	30 Jun 94
	QA Plan	006-QQ-92	8 Sept 94
	Instrument Installation Spec.	00157ST92 Rev. 1	30 Jun 94
	Pre-Test Analyses	RAI 900.35	31 May 94
	Data Acquisition Spec.	0095RS91 Rev. 1	30 Jun 94
	Test Plan and Procedure	0098PP91 Rev. 1	16 Aug 94
	Process & Instrument Drawing	00209DD93 Rev. 4	12 Dec 94
	Data Transmittal Report	00393RP95, Rev. 0	14 Apr 95, 5 May 95
	PANTHERS/PCC Data Analysis Report	00394RA95, Rev. 0	6 Jul 95
PANTHERS/IC	Test Specification	23A6999 Rev. 4	28 Apr 95
	As-Built Drawing Package	Many	21 Jun 95
	Test Plan	00396RI95, Rev. 0	21 Jun 95
	Test Procedures	00395PP05 Rev. 1	15 May 95
	Process & Instrument Drawing	00210DD93 Rev. 3	21 Jun 95
	Apparent Test Results (Phase 1)		
	Pre-Test Analysis Package	MFN 097-95	5 July 95
	Apparent Test Results (Phase 3)		
	Data Transmittal Report		
	PANTHERS/IC Data Analysis Report		

**Table A.2-3 SBWR Analysis
Documentation Submittals**

<p><u>Pre-Test Predictions</u></p> <p>PANTHERS/PCC - Complete [59] PANDA Steady State (S Series) - Complete [56] PANTHERS/IC PANDA M2 PANDA M3 - Complete [79] PANDA M5 PANDA M9 GIRAFFE/Helium (blind post-test) GIRAFFE/SIT (blind post-test)</p>
<p><u>Preliminary Validation Results</u></p> <p>PANDA Steady State Tests PANDA Transient Tests PANTHERS/PCC PANTHERS/IC GIRAFFE/Helium GIRAFFE/SIT</p>

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

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

* % means percent of full-scale

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

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

* Nominal Value

† Superheat conditions are relative to the steam partial pressure.

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

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

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

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

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

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

* Nominal Value

† Superheat referenced to steam partial pressure.

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

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

* Nominal Value

† Superheat referenced to steam partial pressure.

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

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

* Nominal Value

† Superheat referenced to steam partial pressure.

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

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

Table A.3-4 Required Thermodynamic & Hydraulic Measurements - IC Test

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

* % means percent of full-scale

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

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

Table A.3-5b PANTHERS/IC Startup and Operation - Test Matrix

Test Condition Number	No. of Cycles	Test Group Number	Initial Pressure, P1 [MPag(psig)]	Inlet Pressure, P2 [MPag (psig)]	Initial Pool Temp. [(°C °F)]
1	2	I2	9.480 (1375)	8.618 (1250)	<21 (70)

**Table A.3-5c PANTHERS/IC Noncondensable Gas Effects -
Test Matrix**

Test Condition Number	Test Group Number	Initial Inlet Pressure [MPag (psig)]
12	I3	0.48 (70)
13	I3	2.08 (300)

Table A.3-5d PANTHER/IC Water Level Effects - Test Matrix

Test Condition Number	Test Group Number	Initial Inlet Pressure [MPag (psig)]
14	I4	0.48 (70)
15	I4	2.08 (300)

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

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

Table A.3-7 PANDA Instrumentation Summary

Measurement Type	Instrument Type	Number	Total
Temperature	Chromel-alumel-thermocouples	442	493
	Pt100-Resistance thermometers	21	
	Thermistors (TC ref. temp.)	30	
Pressure	Rosemount Model 33051CA transducer	15	21
	Rosemount Model 2088A transducer	3	
	Rosemount Model 1144A transducer	3	
Pressure difference	Rosemount Model 3051CD transducer	14	27
	Rosemount Model 1151DP transducer	13	
Level	Rosemount Model 3051CD transducer	7	18
	Rosemount Model 1151DP transducer	11	
Gas concentration	Oxygen partial pressure probe	8	8
Flow rate	Vortex flow meter	11	15
	Ultrasonic flow meter	3	
	Hot film flowmeter	1	
Fluid phase detector	Conductivity probe	9	9
Electrical power	Wattmeter	6	7
	Electronic totalizer	1	
Total			598

Table A.3-8 Instrumentation Required for Test S1 to S13

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

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

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

PANDA Test No.	Steam Flow (kg/s)	Air Flow (kg/s)	PANTHERS Test Condition No.	GIRAFFE Phase I, Step 1 Test No.
S1	0.195	0	41	2
S2	0.195	0.003	9	4
S3	0.195	0.006	15	6
S4	0.195	0.016	18	8
S5	0.195	0.034*	23	10
S6	0.26	0	43	3
S10	0.195	0.006	15	6
S11	0.195	0.034*	23	10
S12	0.26	0	43	3
S13 [†]	0.26	0		

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

† Test S13 is to be conducted with PCC pool water level at bottom of upper PCC header. All other tests are to be run with normal PCC pool level (4.5m above bottom of pool).

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

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

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

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

Notes:

- (1) Water levels are specified relative to the top of the PANDA heater bundle.
- (2) The nominal DW condition is no water. However, a small amount of spill from the RPV to the DW at the start of the test is acceptable.

Table A.3-10b Initial Conditions for PANDA Test M7

	RPV	Drywell	Wetwell	GDCS	PCC pools
Total Pressure (kPa)	101	101	101	101	101
Air/Nit. Pressure (kPa)	0	(5)	56	81	n/a
Vapor Temperature (K)	373	(5)	352	333	n/a
Liquid Temperature (K)	373	(5)	352	333	373
Collapsed WL (m) ⁽¹⁾	(2)	(3)	3.8	(4)	23.2

Notes:

- (1) Water levels are specified relative to the top of the PANDA heater bundle.
- (2) The RPV water level will be such that the swollen level rises just to the elevation of the steamline when the heaters are ramped to their initial power level.
- (3) The nominal DW condition is no water. However, a small amount of spill from the RPV to the DW at the start of the test is acceptable.
- (4) The GDCS water level will be set in hydrostatic equilibrium with the initial RPV level.
- (5) PANDA capability for producing a dry air environment in the DW is currently being investigated.

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

Level/Ring	Ring 1(R=2.5m)	Ring 2(R=3.0m)	Ring 3(R=4.26m)	Ring 4(R=7.8m)	Ring 5(R=13.75m)	Ring 6(R=15.75m)
Level 9 Z=24.812m	vol=29.598 m ³ dw head p=293.85 kPa pa=101.95 kPa alp=1.000 Tv=394.56 K TI=n/a	vol=13.063 m ³ dw head p=293.85 kPa pa=50.79 kPa alp=1.000 Tv=409.63 K TI=n/a	vol=43.452 m ³ upper dw p=293.85 kPa pa=9.34 kPa alp=1.000 Tv=439.97 K TI=n/a	vol=202.792 m ³ upper dw p=293.85 kPa pa=9.66 kPa alp=1.000 Tv=440.18 K TI=n/a	not used	not used
Level 8 Z=23.30m	vol=85.726 m ³ rpv-strm dome p=294.71 kPa pa=3.91 kPa alp=0.908 (level) Tv=424.40 K TI=404.65 K	vol=37.720 m ³ rpv-strm dome p=294.70 kPa pa=3.10 kPa alp=0.909 (level) Tv=419.81 K TI=405.26 K	vol=196.606 m ³ upper dw p=293.89 kPa pa=9.23 kPa alp=1.000 Tv=439.92 K TI=n/a	vol=917.570 m ³ upper dw p=293.89 kPa pa=8.91 kPa alp=1.000 Tv=437.24 K TI=n/a	vol=232.509 m ³ gdcs pl-up cell p=293.90 kPa pa=48.12 kPa alp=0.785 (level) Tv=417.02 K TI=335.24 K	vol=465.047 m ³ gdcs pl-up cell p=293.90 kPa pa=49.37 kPa alp=0.786 (level) Tv=414.92 K TI=333.41 K
Level 7 Z=19.60m	vol=20.263 m ³ rpv-sep region p=321.90 kPa pa=0.00 kPa alp=0.302 (mixture) Tv=408.09 K TI=408.22 K	vol=50.178 m ³ rpv-up dwncmr p=321.64 kPa pa=0.00 kPa alp=0.017 (mixture) Tv=408.06 K TI=407.88 K	vol=127.528 m ³ upper dw p=293.94 kPa pa=10.75kPa alp=1.000 Tv=433.94 K TI = n/a	vol=595.180 m ³ upper dw p=293.94 kPa pa=9.60 kPa alp=1.000 Tv=437.75 K TI=n/a	vol=150.817 m ³ gdcs pl-low cell p=313.08 kPa pa=n/a alp=0.000 Tv=n/a TI=330.66 K	vol=301.652 m ³ gdcs pl-low cell p=313.04 kPa pa=n/a alp=0.000 Tv=n/a TI=333.41 K
Level 6 Z=17.20m	vol=6.597 m ³ rpv-stnd pipes p=332.29 kPa pa=n/a alp=0.000 Tv=n/a TI=409.15 K	vol=12.493 m ³ rpv-dwncmr p=335.26 kPa pa=n/a alp=0.000 Tv=n/a TI=407.77 K	vol=31.882 m ³ upper dw p=293.97 kPa pa=16.54 kPa alp=1.000 Tv=412.54 K TI=n/a	vol=148.795 m ³ upper dw p=293.97 kPa pa=15.70 kPa alp=1.000 Tv=415.65 K TI=n/a	not used	not used
Level 5 Z=16.60m	vol=93.030 m ³ rpv-chimney p=355.91 kPa pa=n/a alp=0.000 Tv=n/a TI=409.15 K	vol=40.436 m ³ rpv-dwncmr p=358.90 kPa pa=n/a alp=0.000 Tv=n/a TI=407.34 K	vol=244.429 m ³ drywell p=294.01 kPa pa=16.56 kPa alp=1.000 Tv=412.42 K TI=n/a	vol=1140.763 m ³ drywell p=294.01 kPa pa=16.51 kPa alp=1.000 Tv =412.73 K TI=n/a	vol=1852.985 m ³ ww-vap space p=284.79 kPa pa=255.43 kPa alp=1.000 Tv=346.52 K TI=n/a	vol=852.628 m ³ ww-vap space p=284.79 kPa pa=262.35 kPa alp=1.000 Tv=342.06 K TI=n/a
Level 4 Z=12.00m	vol=40.448 m ³ rpv-chimney p=385.89 kPa pa=n/a alp=0.000 Tv=n/a TI=409.13 K	vol=17.624 m ³ rpv-dwncmr p=388.93 kPa pa=n/a alp=0.000 Tv=n/a TI=407.13 K	vol=106.273 m ³ drywell p=294.07 kPa pa=16.55 kPa alp=1.000 Tv=412.40 K TI=n/a	vol=495.984 m ³ drywell p=294.07 kPa pa=16.53 kPa alp=1.000 Tv=412.65 K TI=n/a	vol=805.646 m ³ ww-vap space p=284.88 kPa pa=244.15 kPa alp=0.839 (level) Tv=351.39 K TI=352.31 K	vol=370.708 m ³ ww-vap space p=284.88 kPa pa=248.41 kPa alp=0.838 (level) Tv=348.85 K TI=351.97 K
Level 3 Z=10.00m	vol=44.493 m ³ rpv-chimney p=404.97 kPa pa=n/a alp=0.000 Tv=n/a TI=408.49 K	vol=19.387 m ³ rpv-dwncmr p=408.05 kPa pa=n/a alp=0.000 Tv=n/a TI=406.82 K	vol=33.868 m ³ drywell p=294.10 kPa pa=18.32 kPa alp=1.000 Tv=412.40 K TI=n/a	vol=158.157 m ³ drywell p=294.11 kPa pa=18.30 kPa alp=1.000 Tv=412.53 K TI=n/a	vol=261.432 m ³ ww-sup pool p=298.42 kPa pa=n/a alp=0.000 Tv=n/a TI=352.32 K	vol=120.295 m ³ ww-sup pool p=298.44 kPa pa=n/a alp=0.000 Tv=n/a TI=352.06 K

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

Level/Ring	Ring 1(R=2.5m)	Ring 2(R=3.0m)	Ring 3(R=4.26m)	Ring 4(R=7.8m)	Ring 5(R=13.75m)	Ring 6(R=15.75m)
Level 2 Z=7.80m	vol=18.555 m ³ rpv-core bypass p=429.45 kPa pa=n/a alp=0.000 Tv=n/a TI=404.99 K	vol=18.233 m ³ rpv-dwncmr p=432.37 kPa pa=n/a alp=0.000 Tv=n/a TI=406.36 K	vol=142.033 m ³ drywell p=294.15 kPa pa=16.56 kPa alp=1.000 Tv=412.29 K TI=n/a	vol=26.000 m ³ drywell p=294.15 kPa pa=16.55 kPa alp=1.000 Tv=412.37 K TI=n/a	vol=549.430 m ³ ww-sup pool p=323.73 kPa pa=n/a alp=0.000 Tv=n/a TI=352.03 K	vol=252.814 m ³ ww-sup pool p=323.73 kPa pa=n/a alp=0.000 Tv=n/a TI=351.99 k
Level 1 Z=4.65m	vol=63.912 m ³ rpv-lwr plenum p=466.17 kPa pa=n/a alp=0.000 Tv=n/a TI=403.87 K	vol=18.078 m ³ rpv-lwr plenum p=466.18 kPa pa=n/a alp=0.000 Tv=n/a TI=405.85 K	vol=31.804 m ³ lower dw p=294.22 kPa pa=16.54 kPa alp=1.000 Tv=412.30 K TI=n/a	vol=148.433 m ³ lower dw p=294.22 kPa pa=16.53 kPa alp=1.000 Tv=412.3 K TI=n/a	not used	not used
TEE35	vol=187.2 m ³ (cell 3) lwr dw, Z=-3.4m p=294.29 kPa pa=12.87 kPa alp=1.000 Tv=410.8 K TI=n/a	vol=233.0 m ³ (cell 2) lwr dw, Z=-6.8m p=294.35 kPa pa=9.86 kPa alp=1.000 Tv=408.4 K TI=n/a	vol=213.0 m ³ (cell 1) lwr dw, Z=-10.0m p=305.89 kPa pa=10.94 kPa alp=0.189 (level) Tv=405.1 K TI=405.1 K			

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

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

Table A.3-13 PANDA TRACG Analysis Cases

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

Table A.3-13 PANDA TRACG Analysis Cases

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

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

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

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

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

Notes:

- (1) Suffix "A" in Test Number signifies a repeat test.
- (2) Collapsed water level relative to bottom of RPV.
- (3) Time since reactor scram in SBW. Used to determine decay heat.

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

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

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

Test No.	Test Group	Steam Flow Rate (kg/s)	Nitrogen Partial Pressure (fraction of total press.)	Pressure (kPa)
1	Phase 1	0.02	0	300
2	Phase 1	0.03	0	300
3	Phase 1	0.04	0	300
4	Phase 1	0.03	0.01	300
5	Phase 1	0.02	0.02	300
6	Phase 1	0.03	0.02	300
7	Phase 1	0.04	0.02	300
8	Phase 1	0.03	0.05	300
9	Phase 1	0.02	0.10	300
10	Phase 1	0.03	0.10	300
11	Phase 1	0.04	0.10	300
12	Phase 1	0.03	0.02	200
13	Phase 1	0.03	0.02	400

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

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

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

Parameter	Value	Tolerance
RPV Pressure (kPa)	295	± 6 kPa
Initial Heater Power (kW)	93	± 1 kW
RPV Water Level (m)*	12.0	± 0.150 m
Drywell Pressure (kPa)	294	± 4 kPa
Wetwell Pressure (kPa)	285	± 4 kPa
Wetwell Nitrogen Pressure (kPa)	240	± 4 kPa
GDCS Gas Space Pressure (kPa)	294	± 4 kPa
GDCS Nitrogen Pressure (kPa)	274	± 4 kPa
Suppression Pool Temperature (K)	352	± 2 K
PCC Pool Temperature (K)	373	± 2 K
GDCS Pool Temperature (K)	333	± 2 K
GDCS Pool Level * (m)	†	
Suppression Pool Level* (m)	3.25	± 0.075 m
PCC Pool Collapsed Water Level* (m)	23.2	± 0.075 m
PCC Vent Line Submergence (m)	0.95	± 0.075 m

* Referenced to the Top of Active Fuel (TAF)

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

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

Parameter	Value	Tolerance
RPV Pressure (kPa)	189	±6 kPa
RPV Collapsed Water Level (m)*	9.1	±0.150 m
Initial Heater Power (kW)	96	±1 kW
Drywell Total Pressure (kPa)	188	±4 kPa
Drywell Nitrogen Partial Pressure(kPa)	53	±4 kPa
Drywell Steam Partial Pressure (kPa)	135	±4 kPa
Wetwell Pressure (kPa)	174	±4 kPa
Wetwell Nitrogen Pressure (kPa)	164	±4 kPa
GDCS Pool Gas Space Total Pressure (kPa)	188	±4 kPa
GDCS Pool Gas Space Nitrogen Partial Pressure(kPa)	151	±4 kPa
Suppression Pool Temperature (K)	326	±2 K
PCC Pool Temperature (K)	373	±2 K
GDCS Pool Temperature (K)	350	±2 K
GDCS Pool Level* (m)	14.1	±0.075 m
Suppression Pool Level *(m)	3.2	±0.075 m
PCC Pool Collapsed Water Level * (m)	23.2	±0.075 m
PCC Vent Line Submergence (m)	0.90	±0.075 m

* Referenced to the TAF.

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

Parameter	Value	Tolerance
RPV Pressure (kPa)	267	±6 kPa
Initial Heater Power (kW)	93	±1 Kw
RPV Water Level (m)*	12.0	±0.150 m
Drywell Pressure (kPa)	266	±4 kPa
Drywell Nitrogen Pressure (kPa)	38	±4 kPa
Wetwell Pressure (kPa)	257	±4 kPa
Wetwell Nitrogen Pressure (kPa)	212	±4 kPa
GDCS Gas Space Pressure (kPa)	266	±4 kPa
GDCS Nitrogen Pressure (kPa)	246	±4 kPa
Suppression Pool Temperature (K)	352	±2 K
PCC Pool Temperature (K)	373	±2 K
GDCS Pool Temperature (K)	333	±2 K
GDCS Pool Level (m)	†	
Suppression Pool Level* (m)	3.25	±0.075 m
PCC Collapsed Water Level *(m)	23.2	±0.075 m
PCC Vent Line Submergence (m)	0.95	±0.075 m

* Referenced to the TAF.

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

Table A.3-21 GIRAFFE/SIT Test Matrix

Test	Break	Single Failure	IC/PCCS on?
GS1	GDL	DPV	No
GS2	GDL	DPV	Yes
GS3	BDL	DPV	Yes
GS4	GDL	GDCS	Yes
GDL = Gravity Drain Line BDL = Bottom Drain Line DPV = Depressurization Valve GDCS = GDCS Injection Valve			

Table A.3-22 Test GS1 Initial Conditions

The information previously provided in this table is in the Test Specification.

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

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

Table A.3-24 PANTHERS PCC Structural Instrumentation

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

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

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

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

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

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

Table A.3-27 PANTHERS IC Structural Instrumentation

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

Table A.3-28 PANTHERS IC Component Demonstration Test Matrix

Test Cond. No.	No. of Cycles	Cycle Type	Initial Inlet Pressure, P1 [MPag (psig)]	Inlet Pressure, P2 [MPag (psig)]	Inlet Pressure, P3 [MPag (psig)]	Initial Pool Temp. °C(°F)
1	1	2	9.480 (1375)	8.618 (1250)	N/A	<21 (70)
16	20	5	8.618 (1250)	8.618 (1250)	N/A	<32 (90)
17	5	6	8.618 (1250)	N/A	N/A	<32 (90)
18	1	7	8.618 (1250)	9.480 (1375)	8.618 (1250)	<32 (90)

Table A.3-29 Comparison of Non-Dimensional Parameters Between SBWR and CRIEPI

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

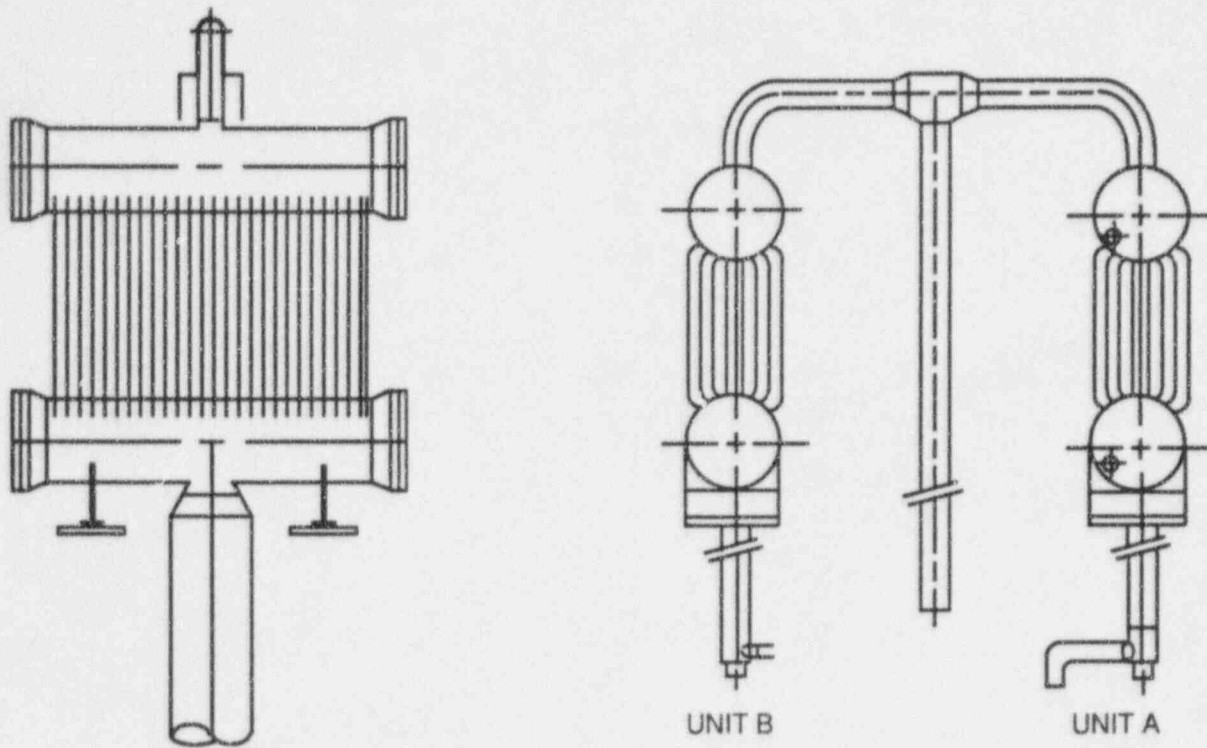


Figure A.3-1 Passive Containment Cooler Test Article

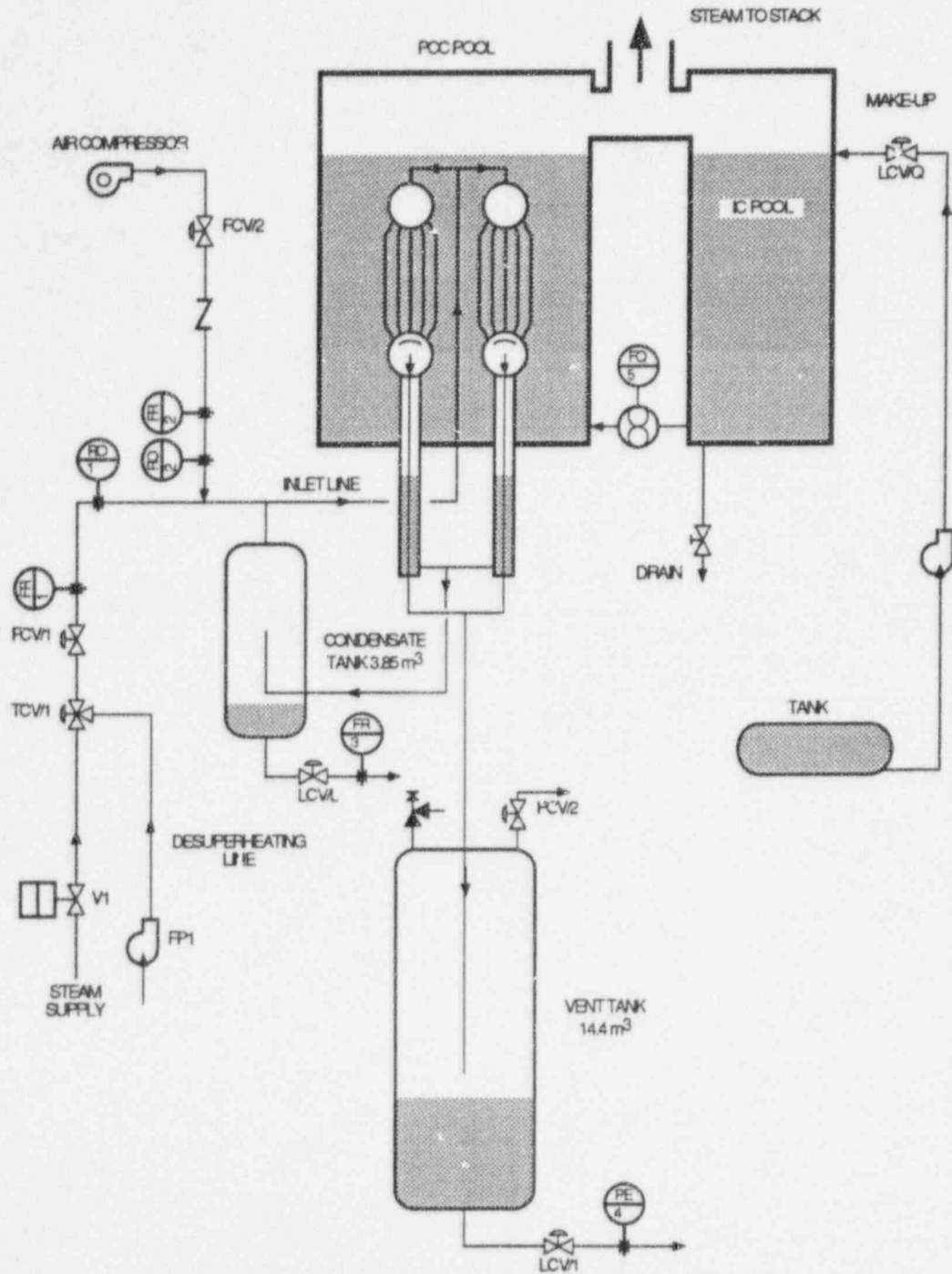


Figure A.3-2 PANTHERS/PCC Test Facility Schematic

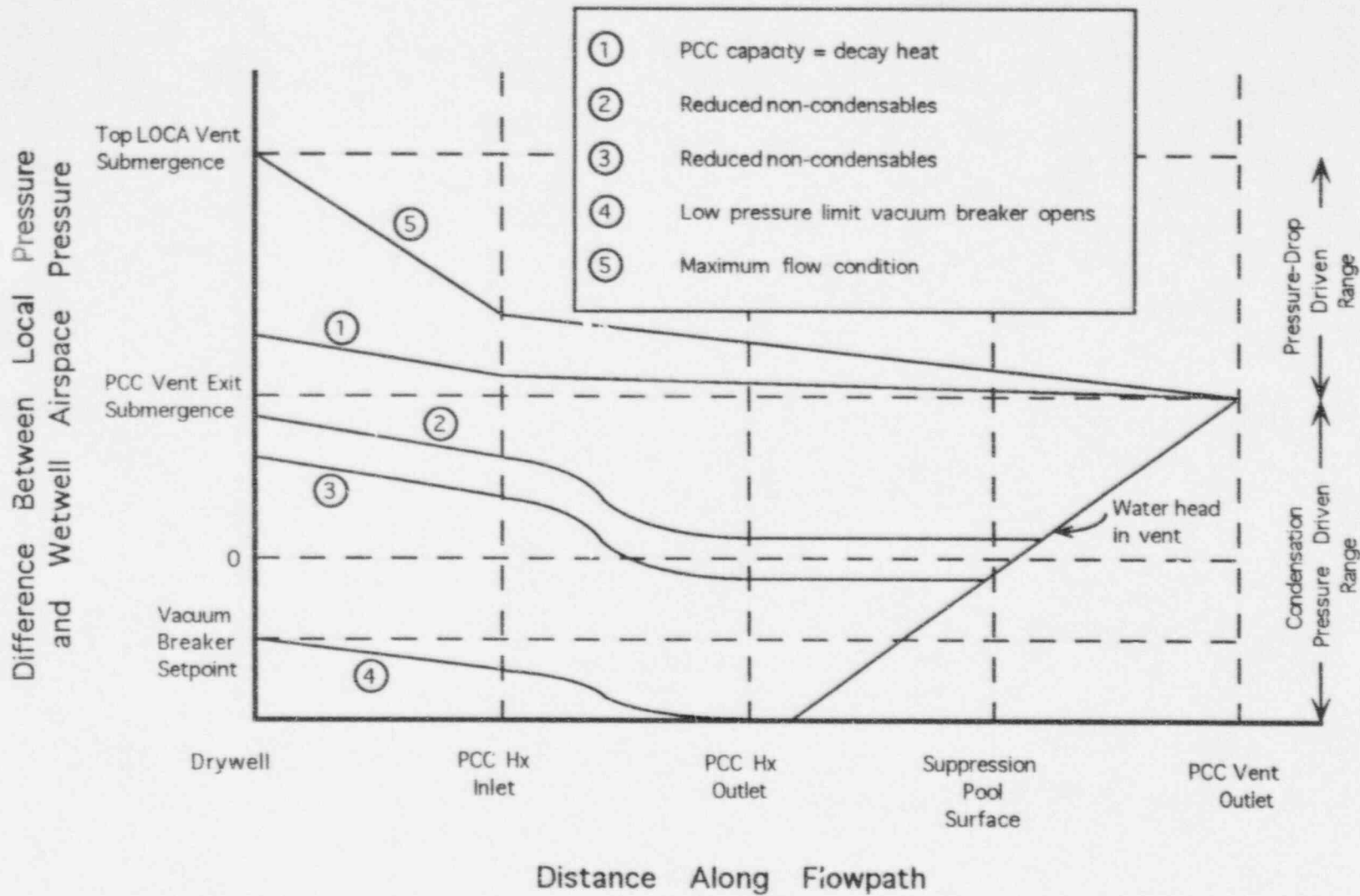


Figure A.3.3 PCC Heat Exchanger Operational Modes

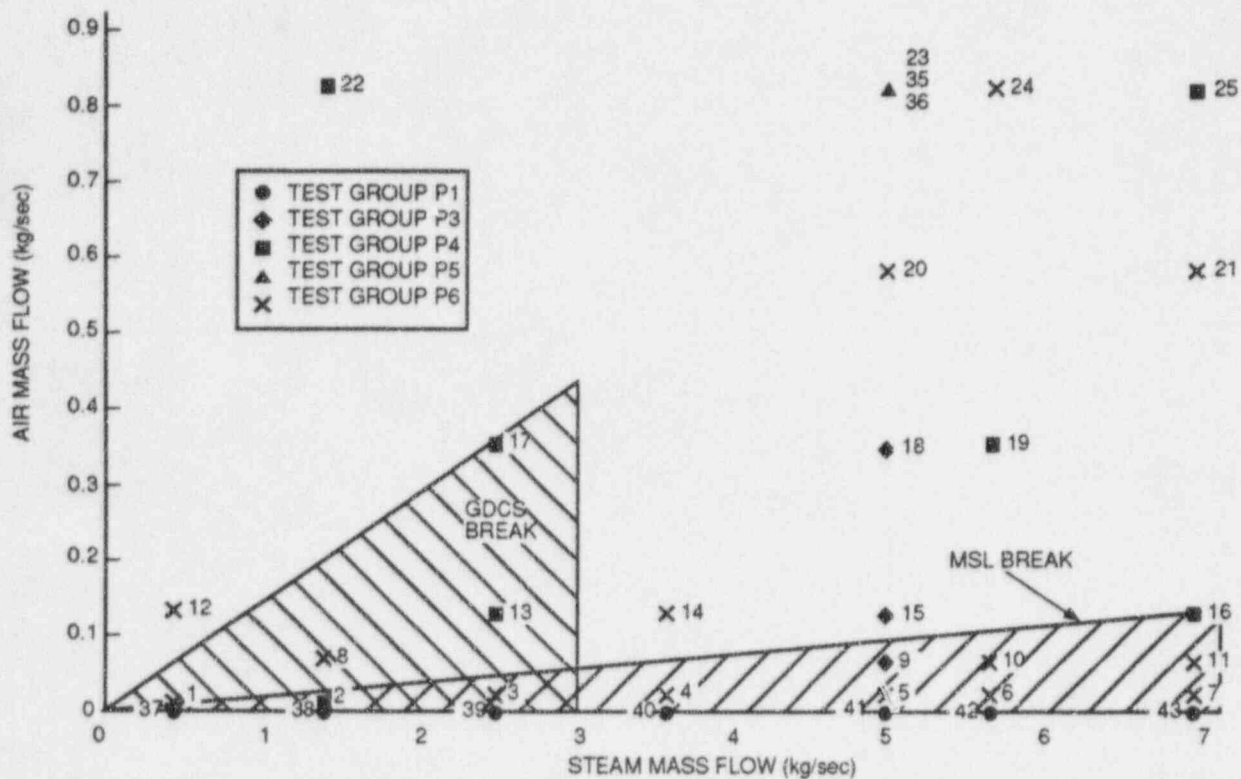


Figure A.3-4 Comparison of PANTHERS/PCC Steam-Air Test Range to SBWR Conditions

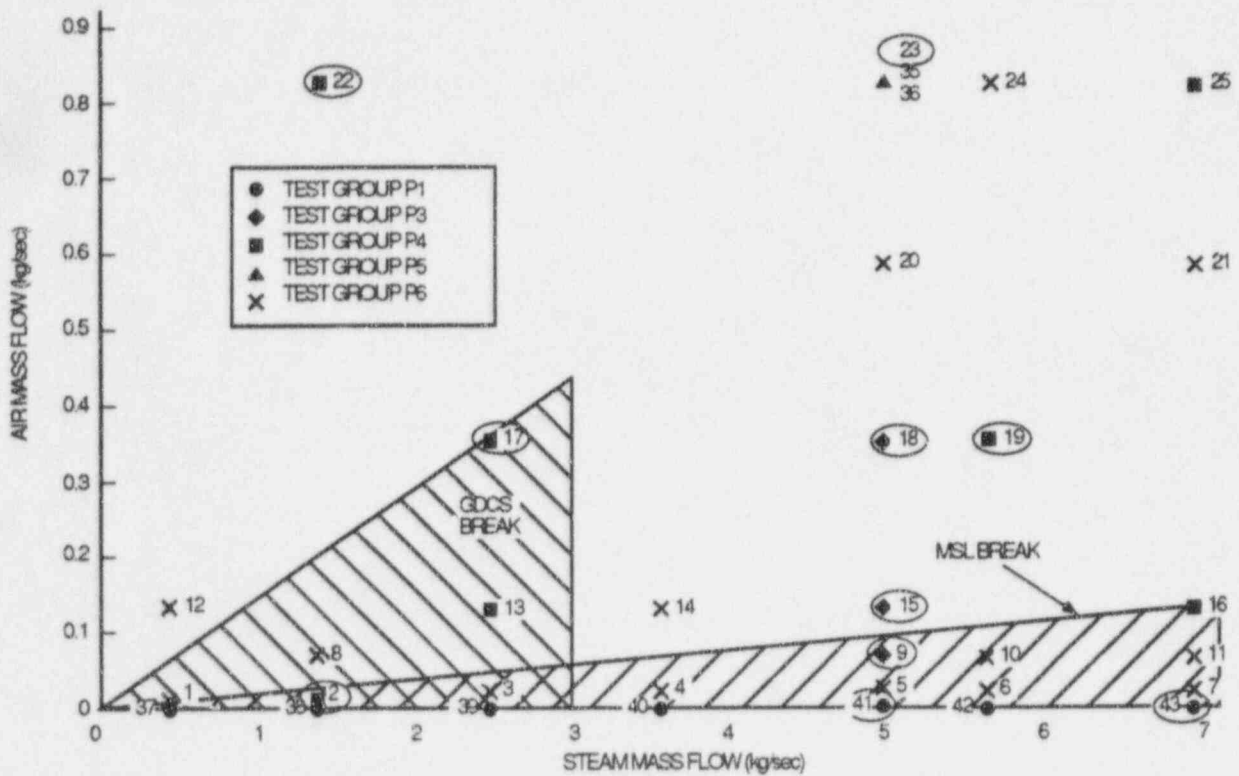


Figure A.3-5 TRACG PANTHERS/PCC Qualification Points

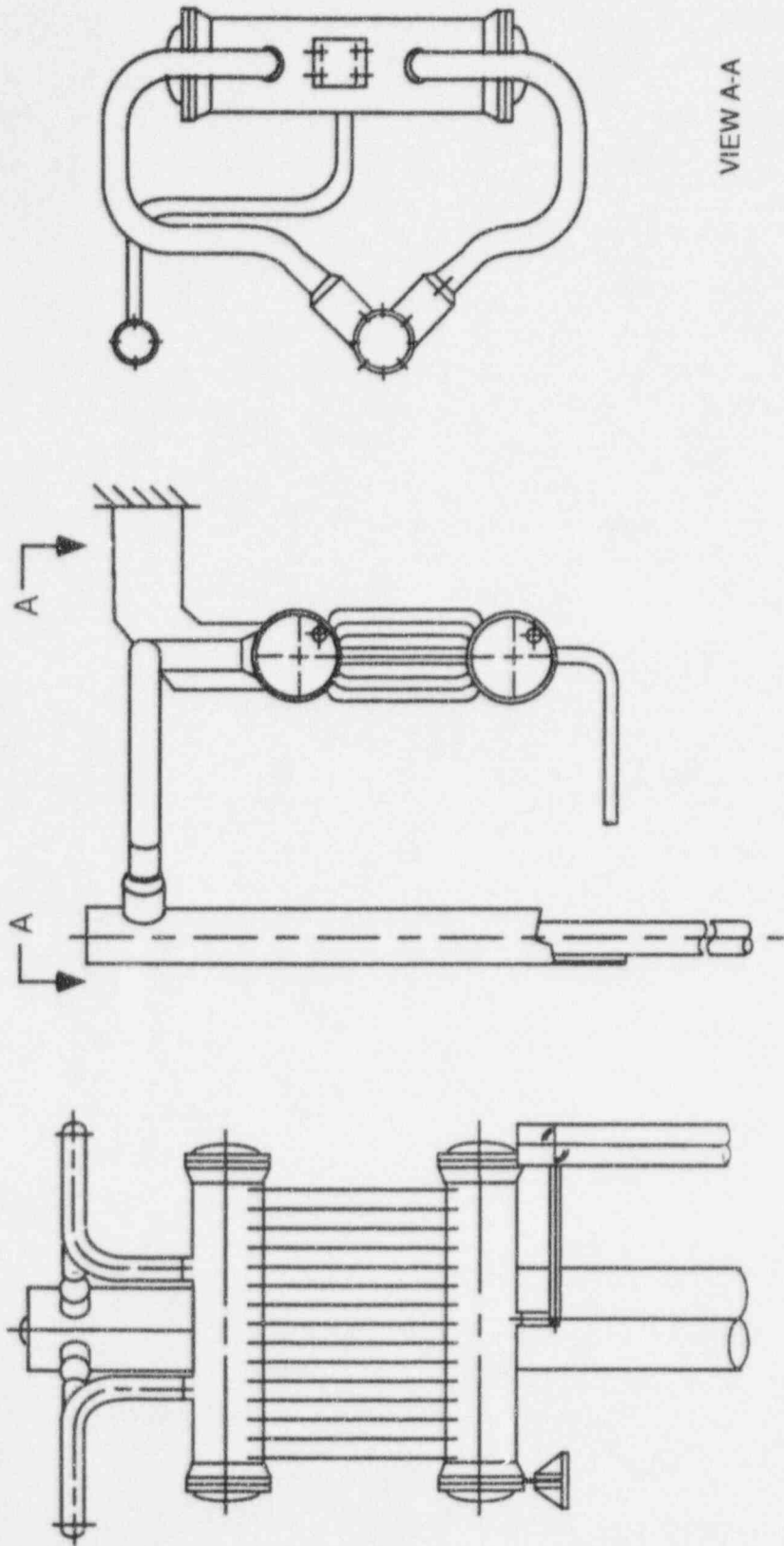


Figure A.3-6 Isolation Condenser Test Article

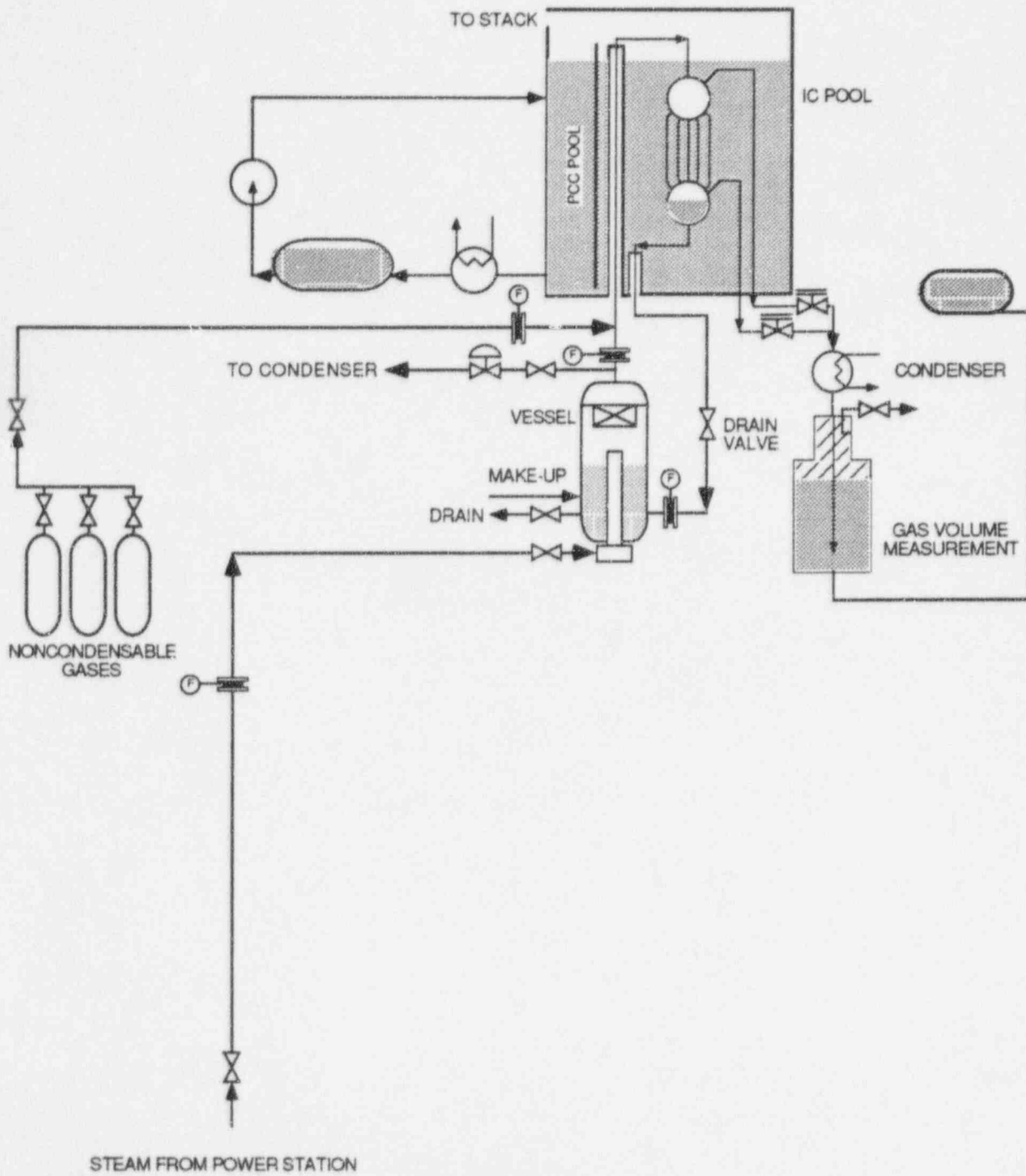


Figure A.3-7 PANTHERS/IC Test Facility Process Diagram

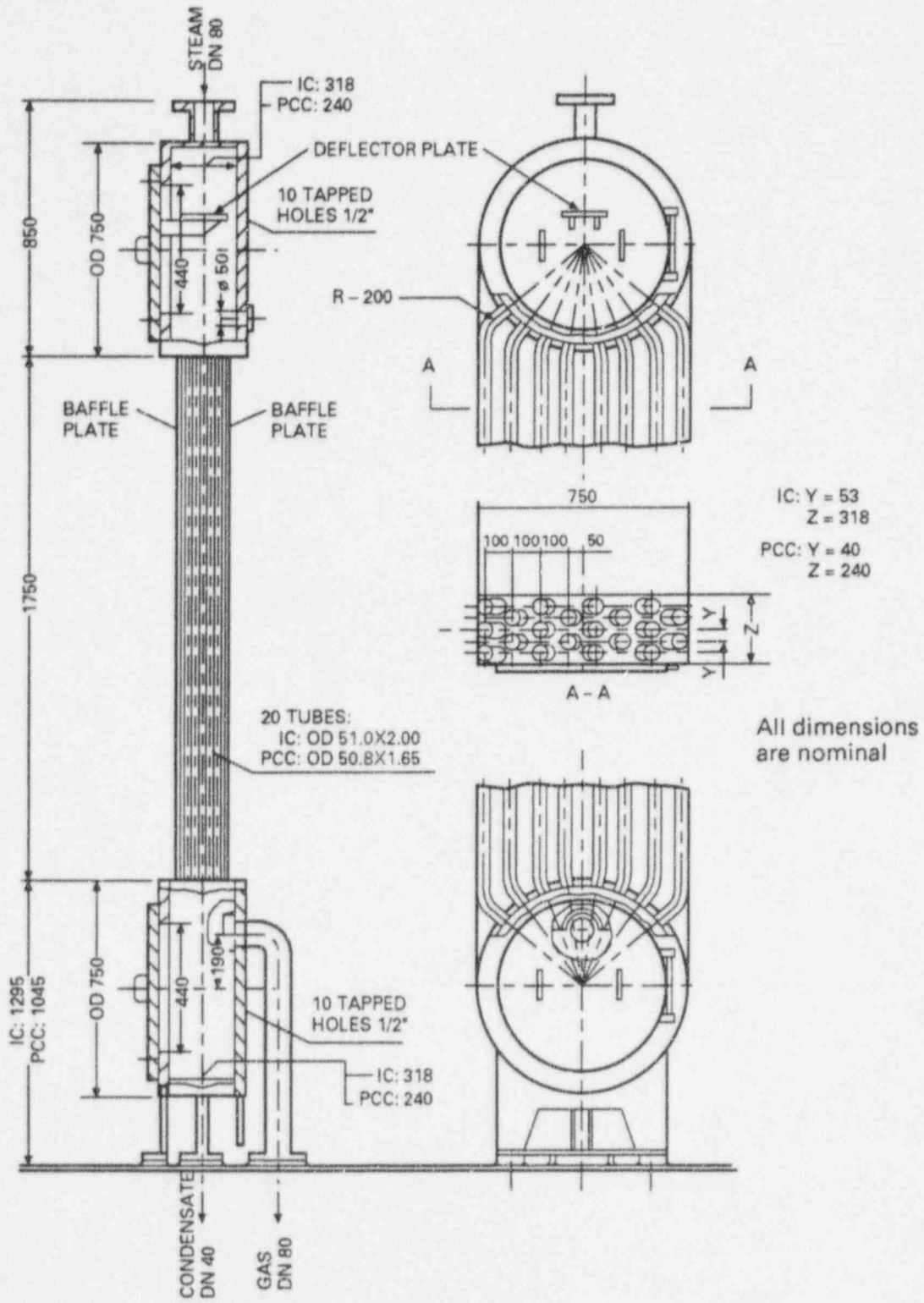


Figure A.3-8 PANDA Facility: IC/PCC Test Units

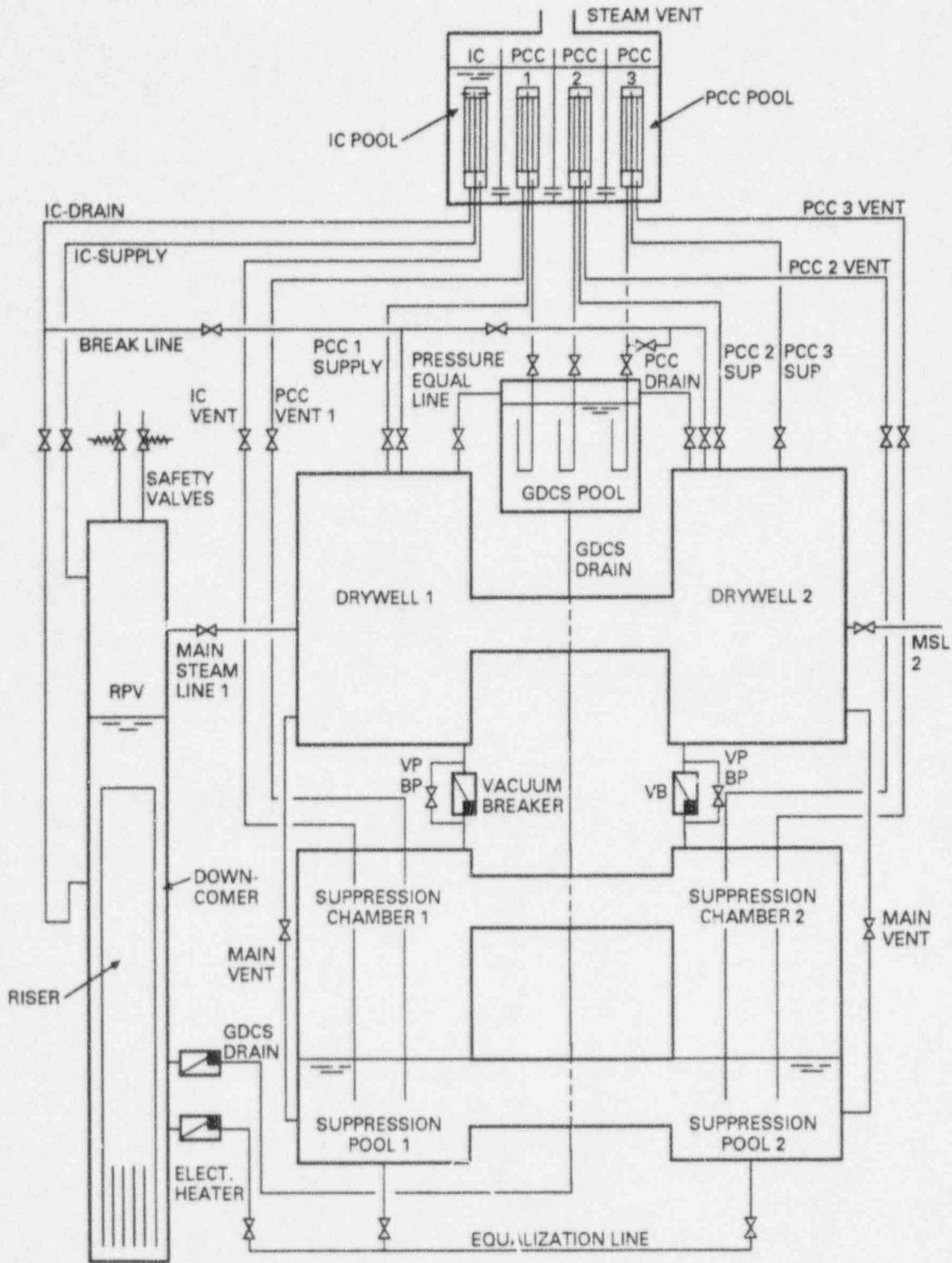


Figure A.3-9 PANDA Facility Schematic

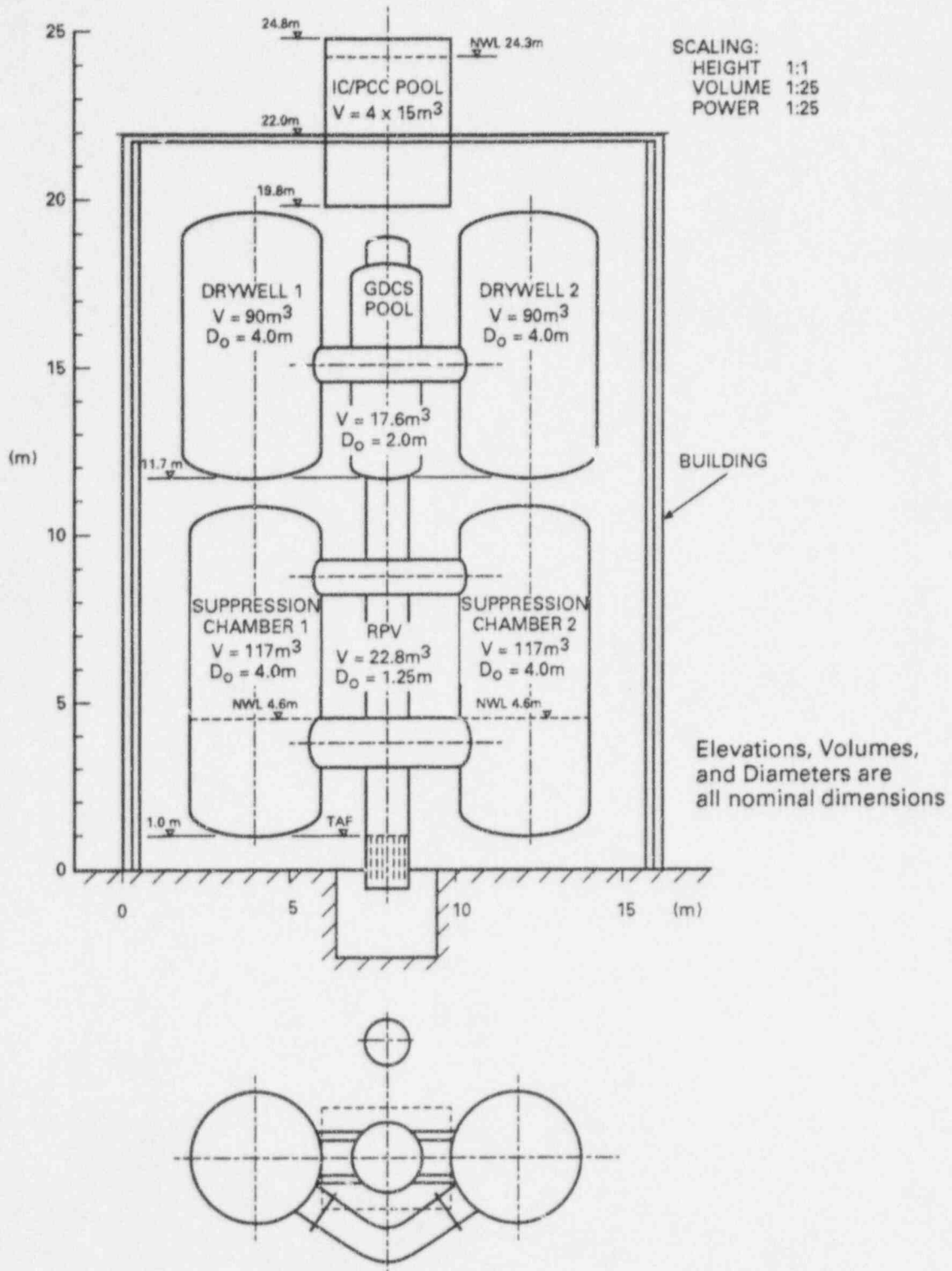


Figure A.3-10 PANDA Facility: Configuration of Vessels

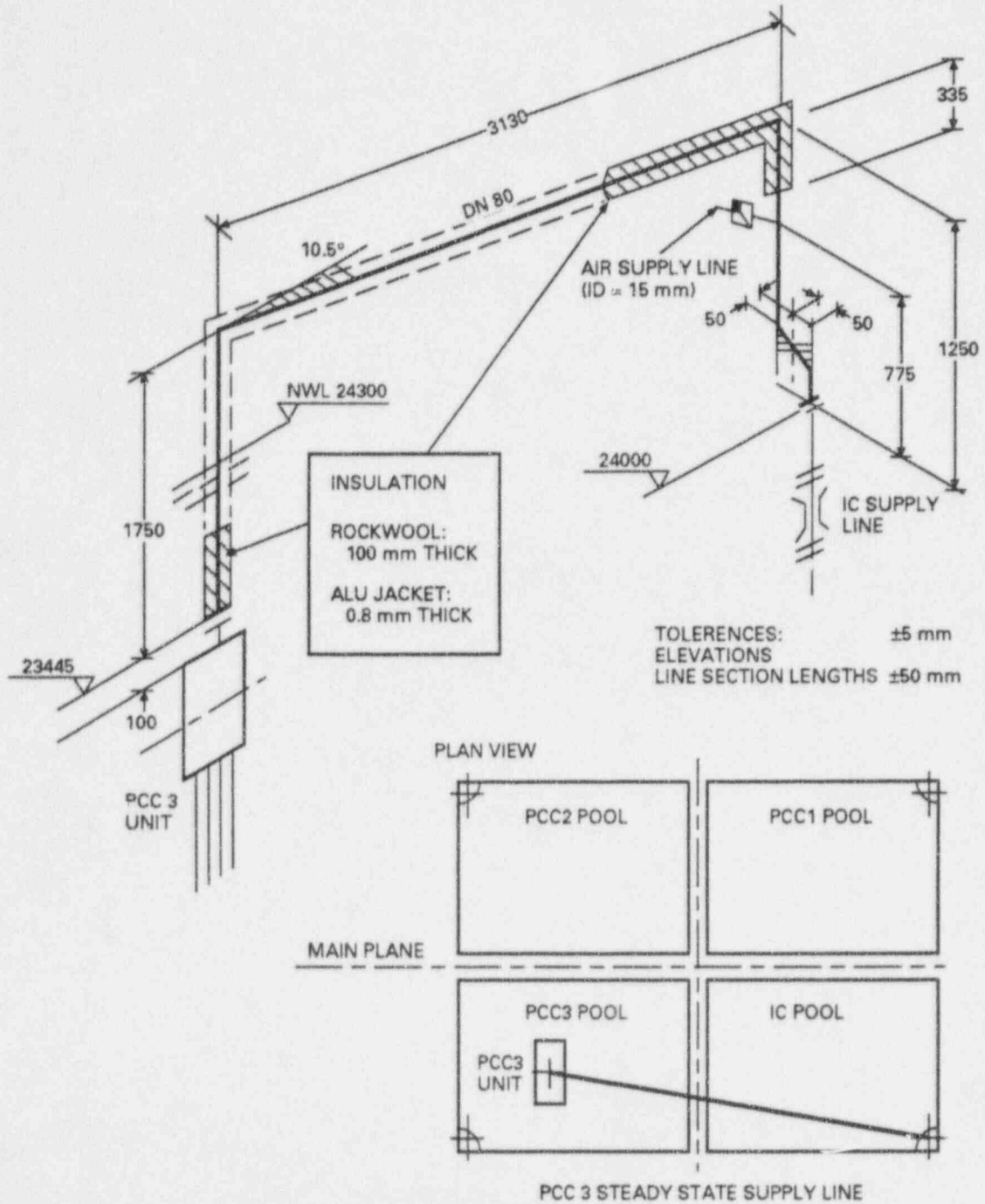


Figure A.3-11 PANDA Facility: PCC 3 Steady State Supply Line

Figure A.3-12 Deleted

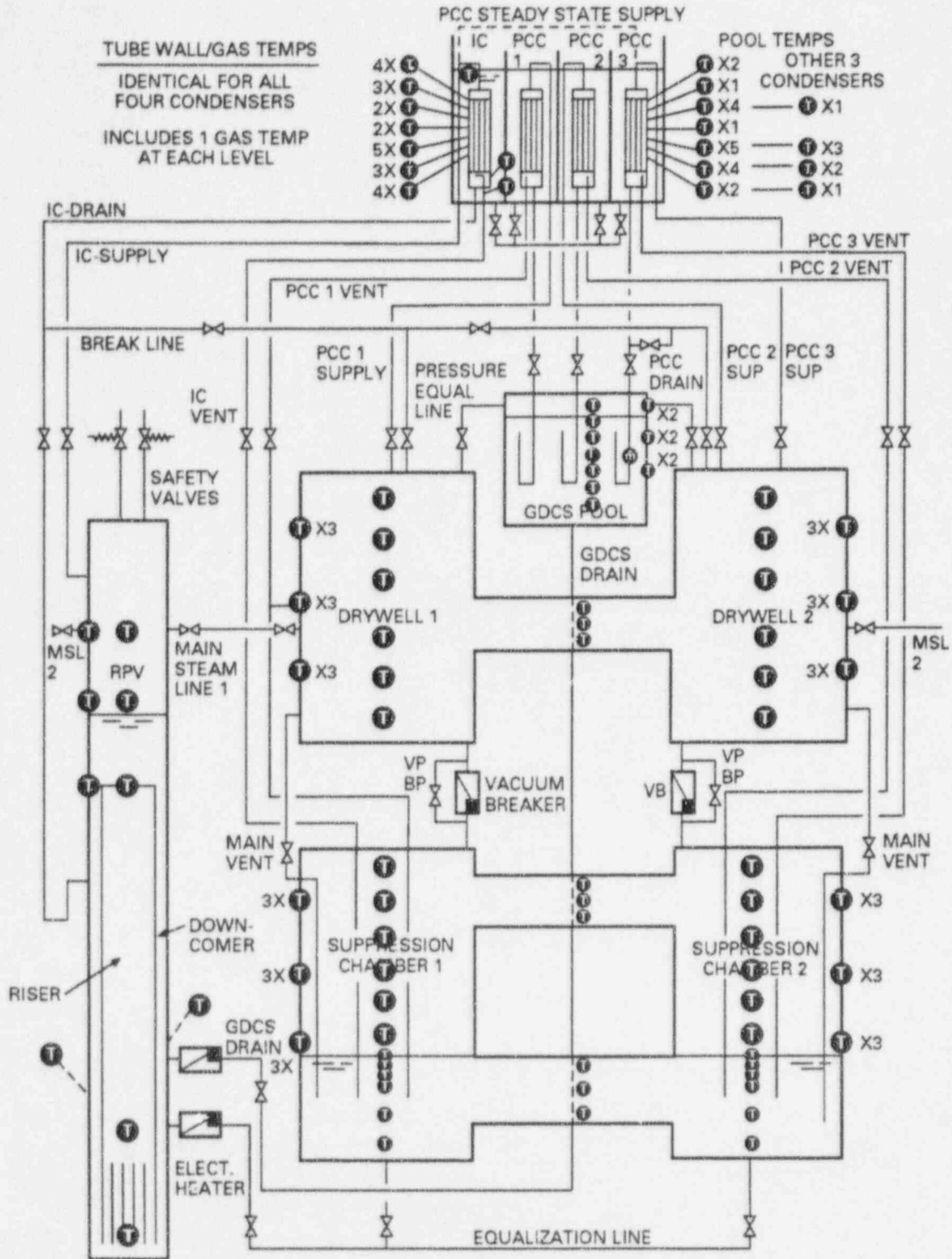


Figure A.3-13a PANDA Instrumentation: Condenser, Pool, and Vessel Temperatures

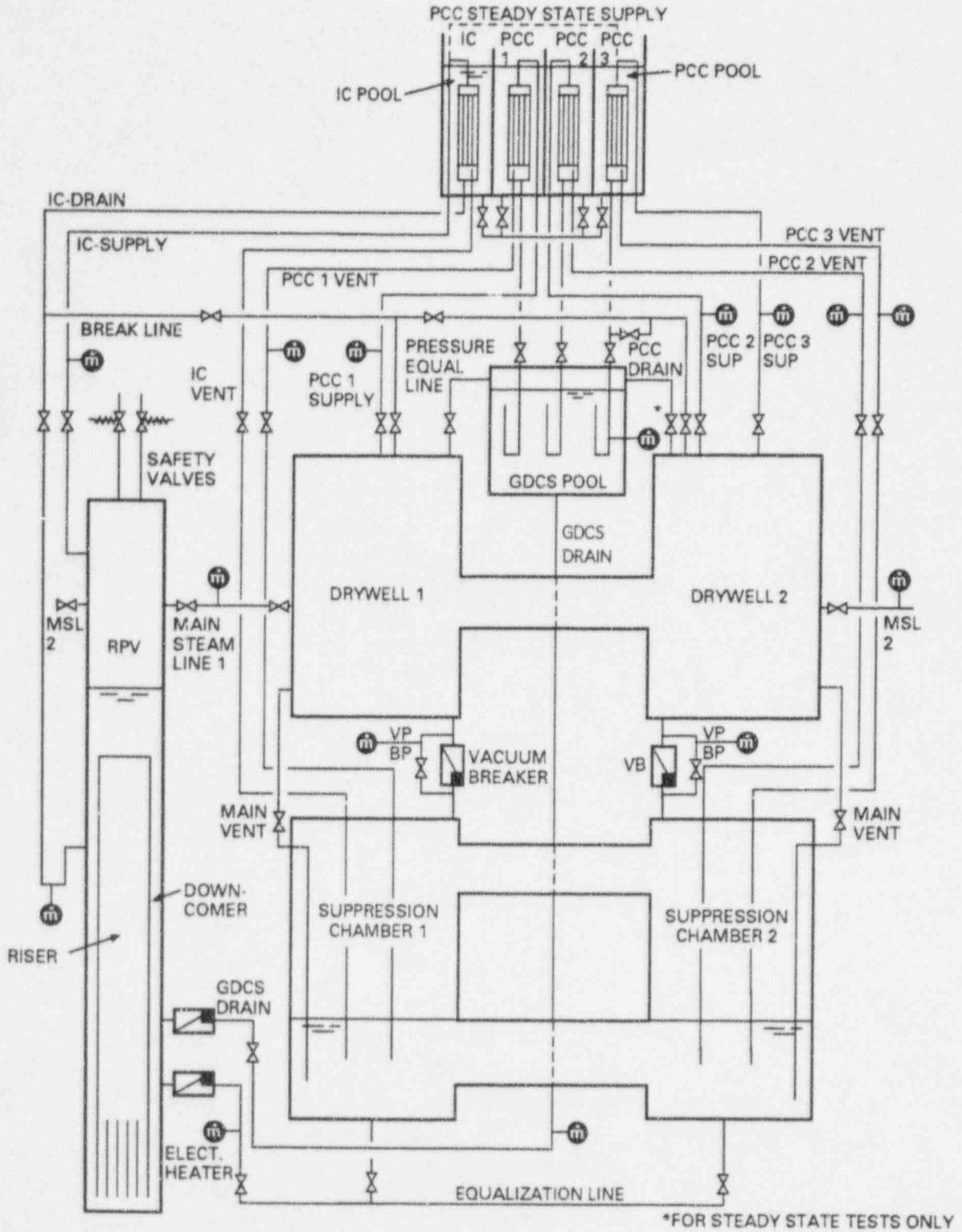


Figure A.3-13b. PANDA Instrumentation: Mass Flow Rates

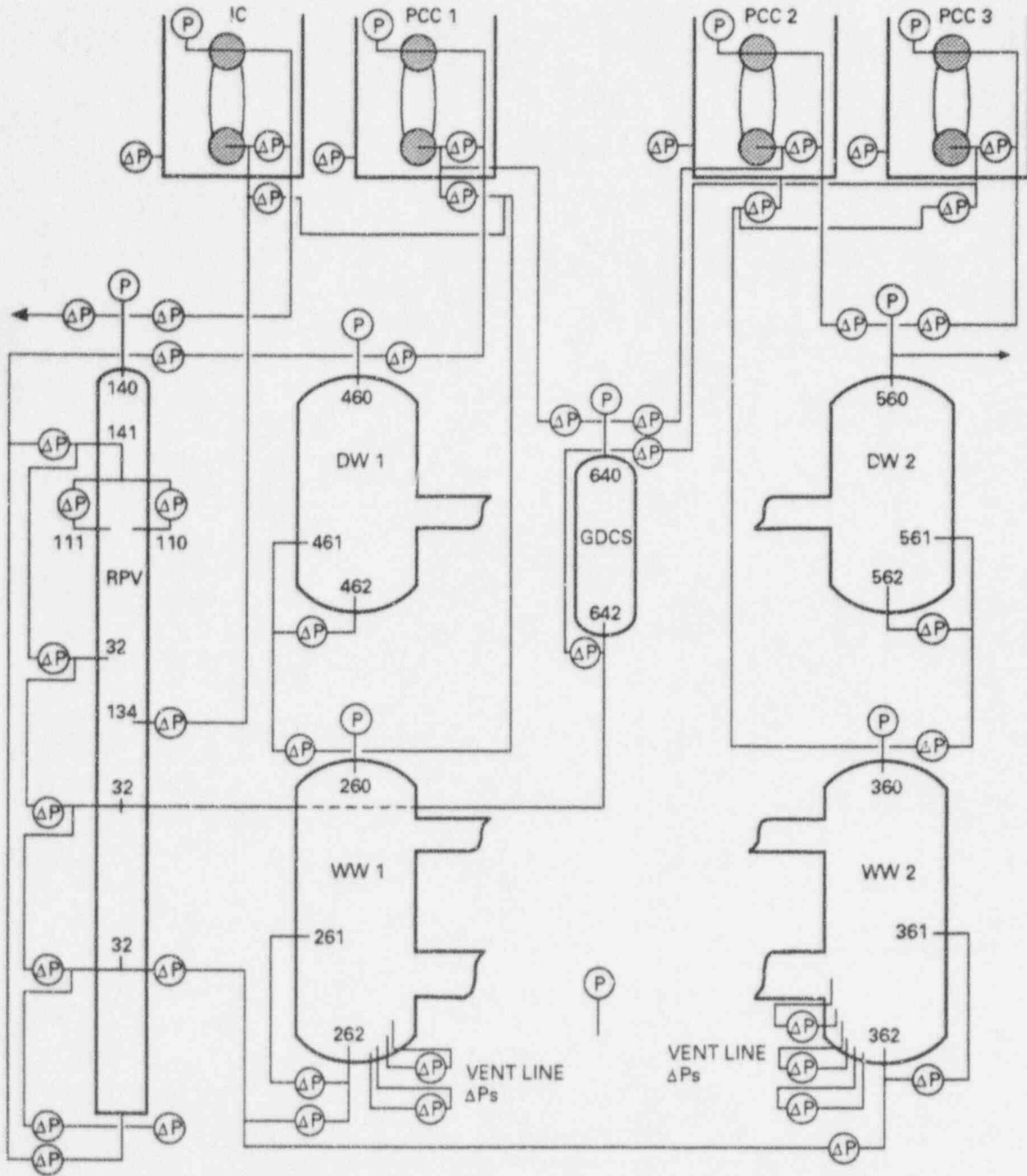


Figure A.3-13c PANDA Instrumentation: Absolute and Differential Pressures

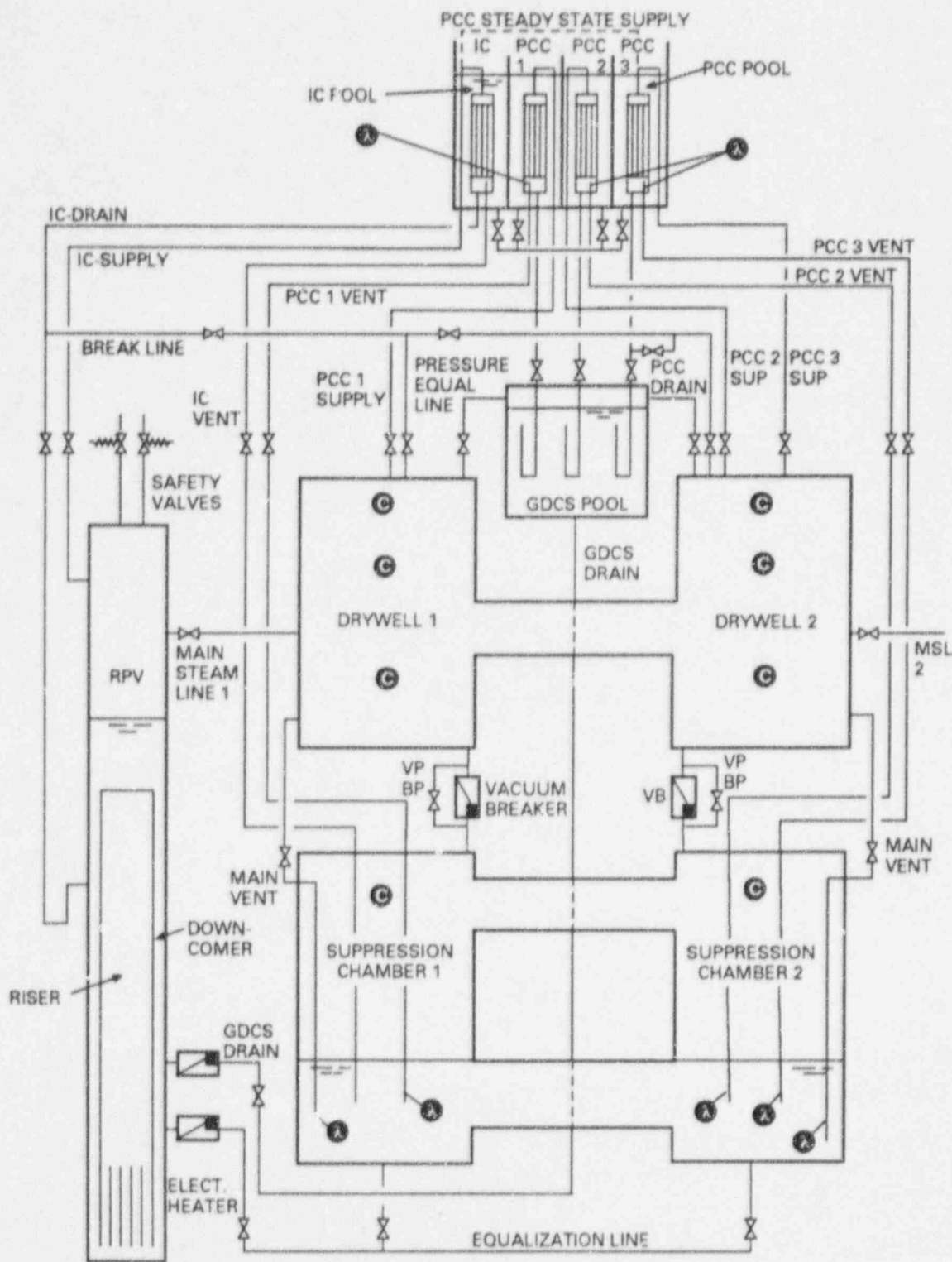


Figure A.3-13d PANDA Instrumentation: Noncondensibles (c) and Phase Detectors (λ)

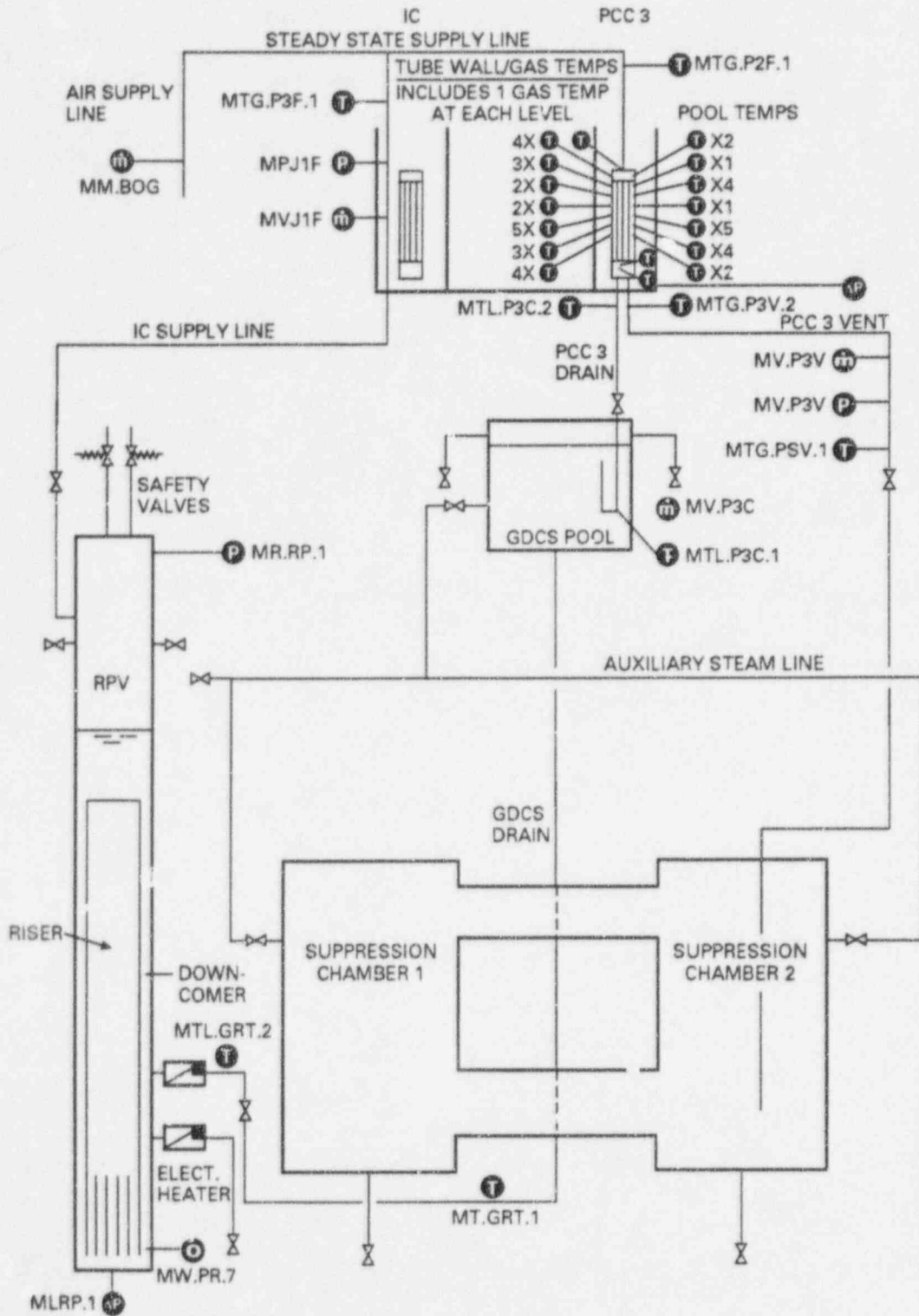


Figure A.3-14 PANDA Steady State Test Instrumentation/Configuration

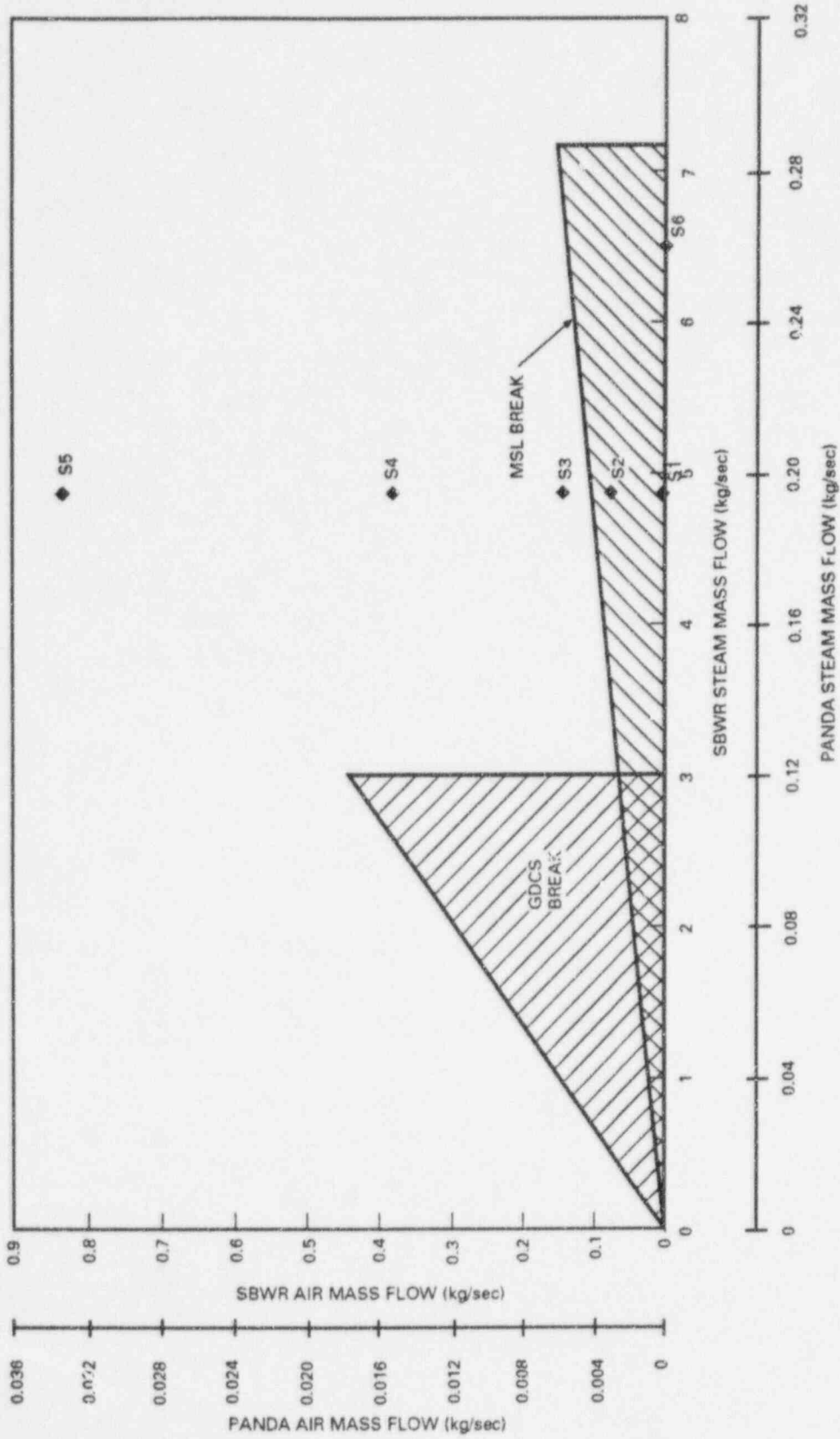


Figure A.3-15 Comparison of PANDA Steady State Test Range to SBWR Condition

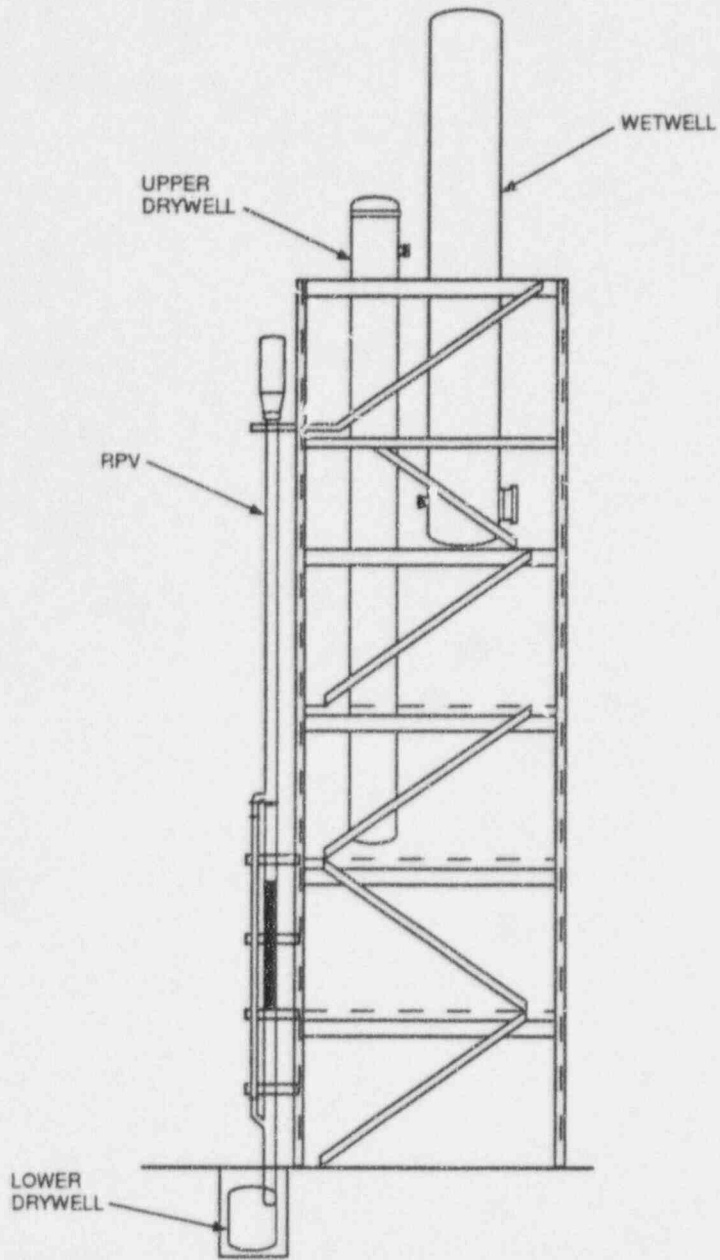


Figure A.3-16 GIST Facility Schematic

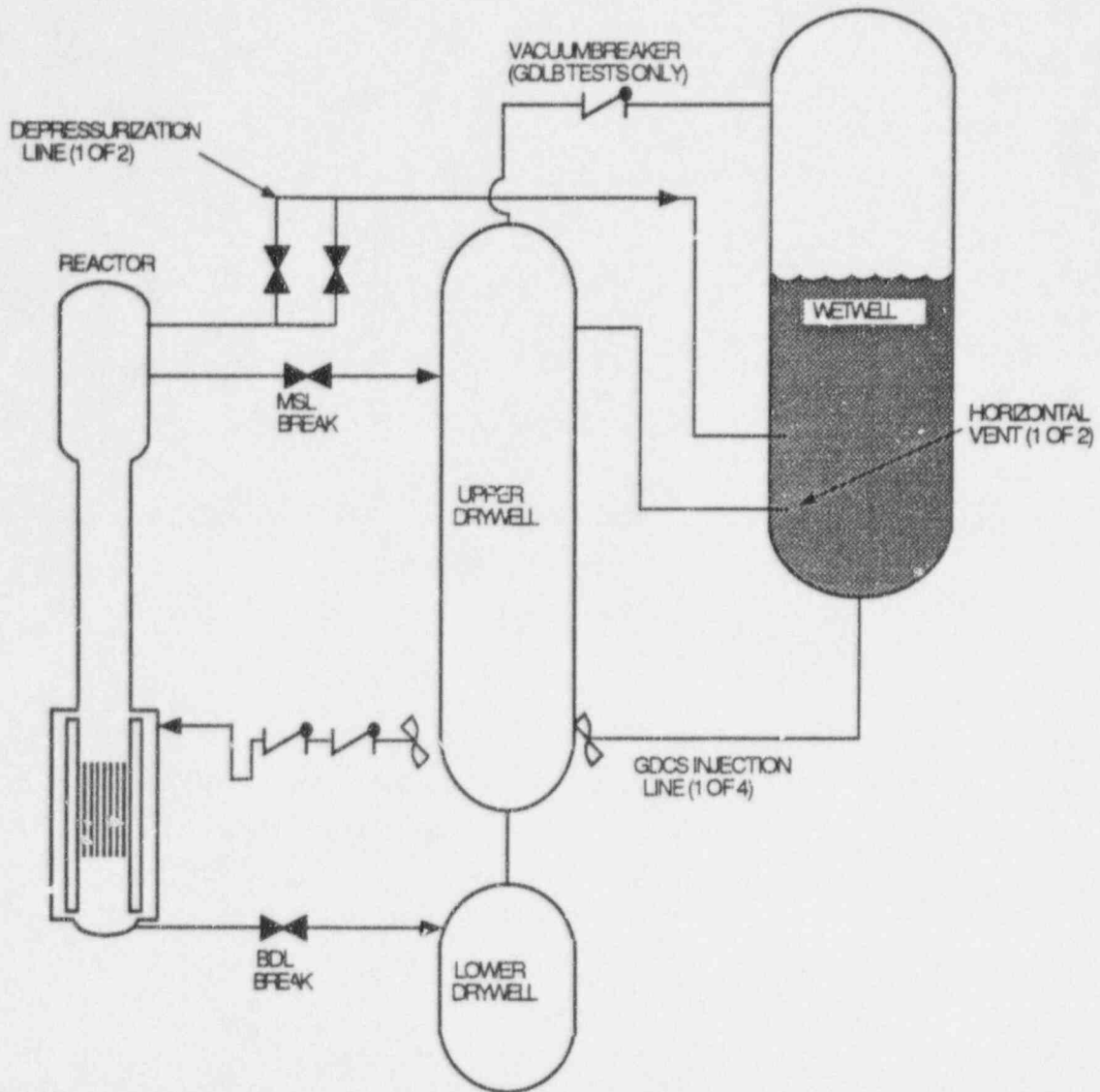


Figure A.3-17 GIST Facility Piping Arrangement

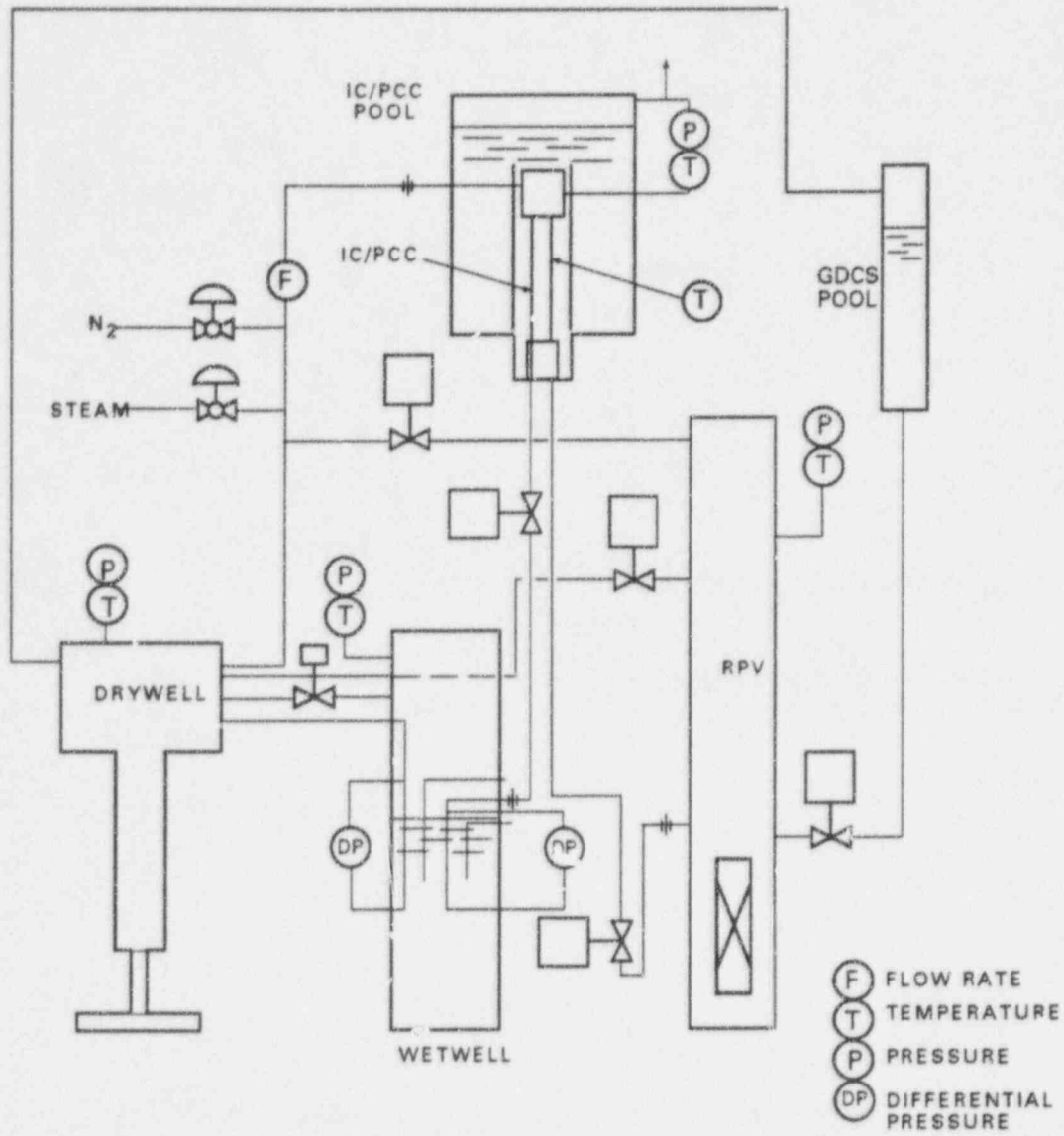


Figure A.3-18 GIRAFFE Test Facility Schematic (Phase 1)

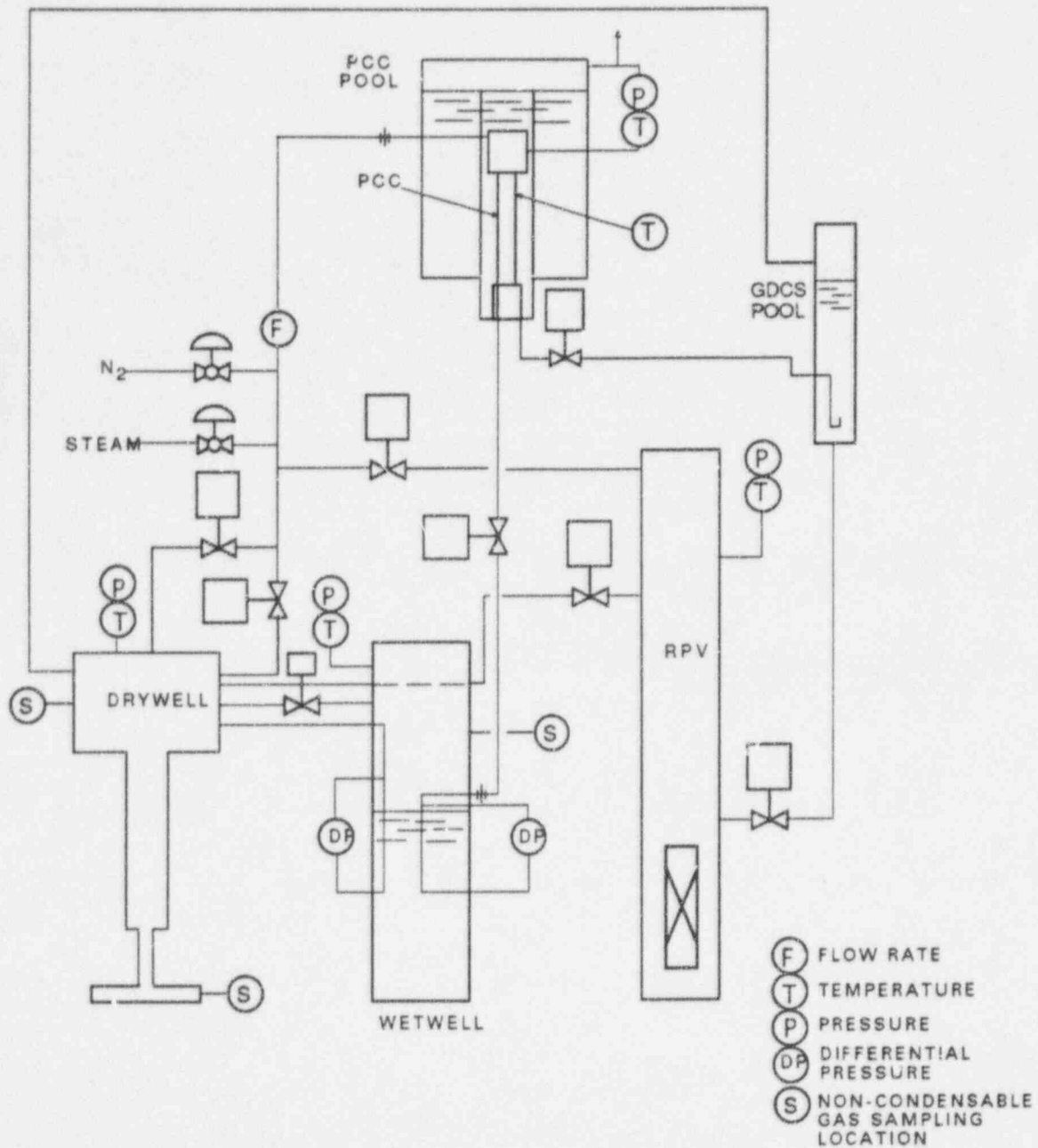


Figure A.3-20 GIRAFFE Test Facility Schematic (Post Phase 1)

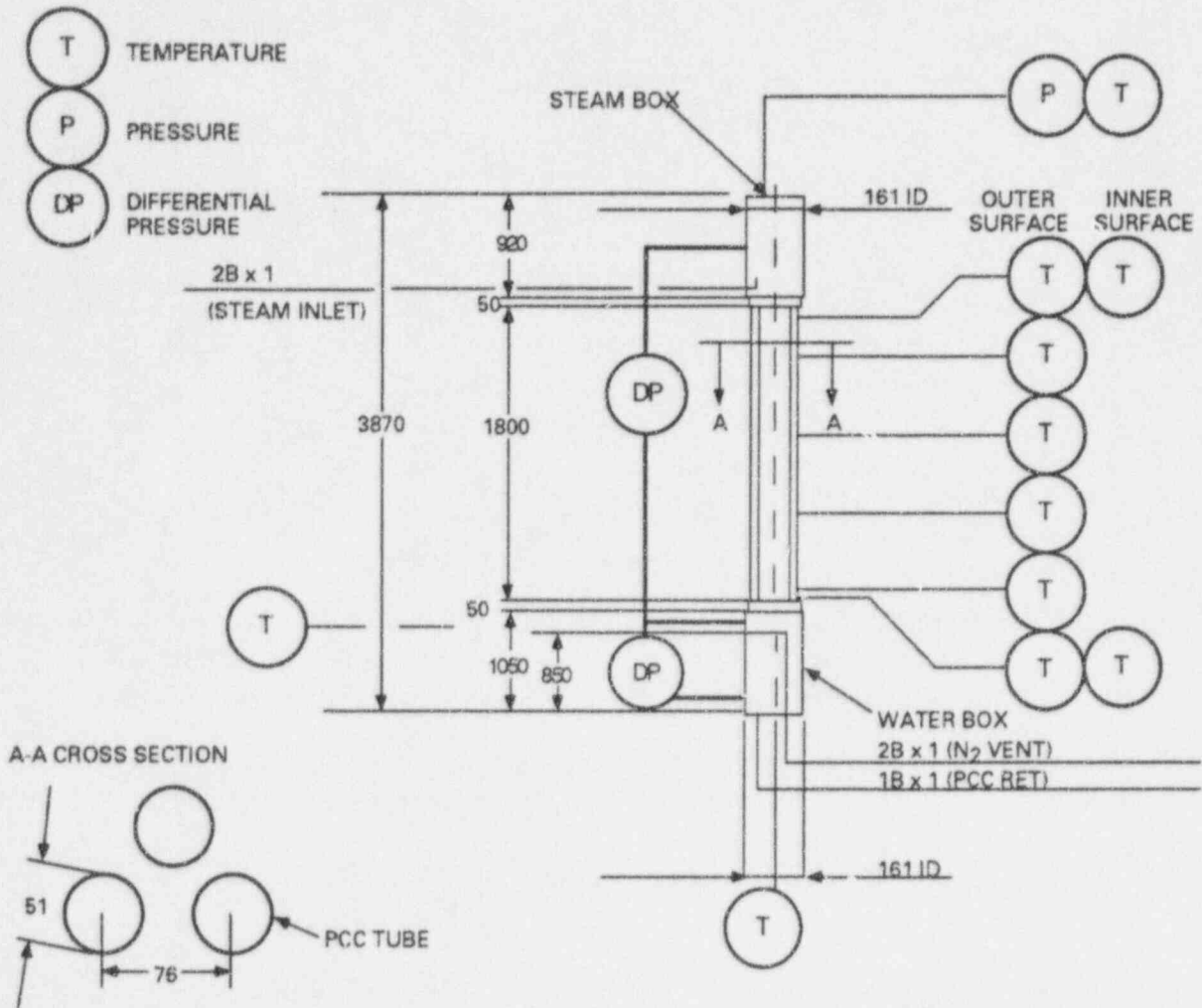
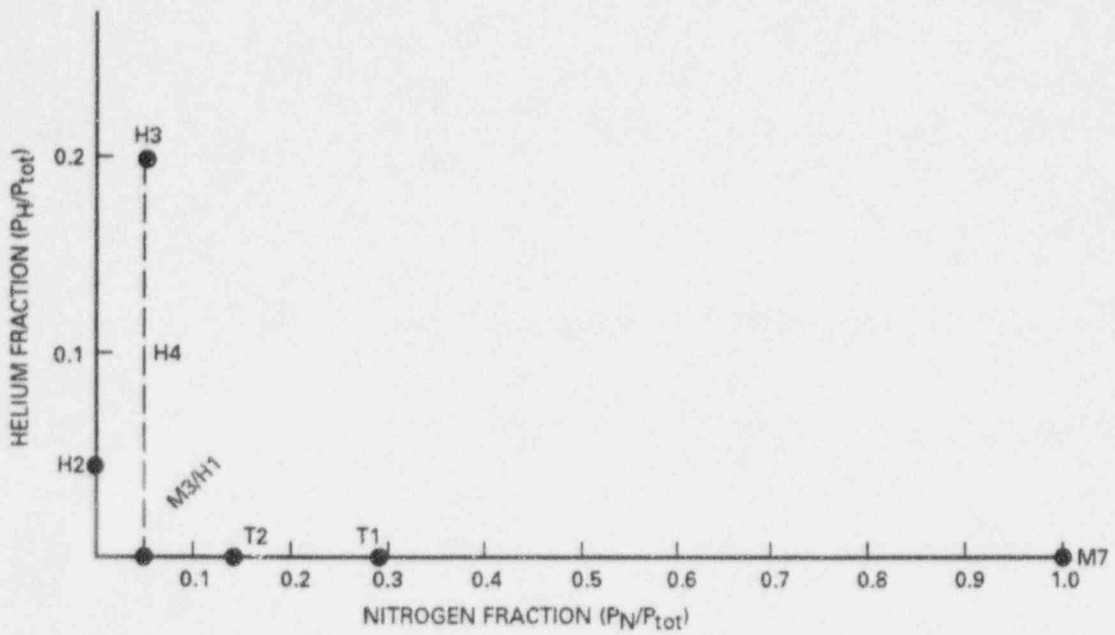


Figure A.3-21 GIRAFFE PCC Unit (Shortened Tubes)



NOTE: TEST H4 HELIUM FRACTION IS UNCONTROLLED.
THE DOTTED LINE SHOWS POTENTIAL BOUNDS.

Figure A.3-22 PCC Startup Initial Condition Map

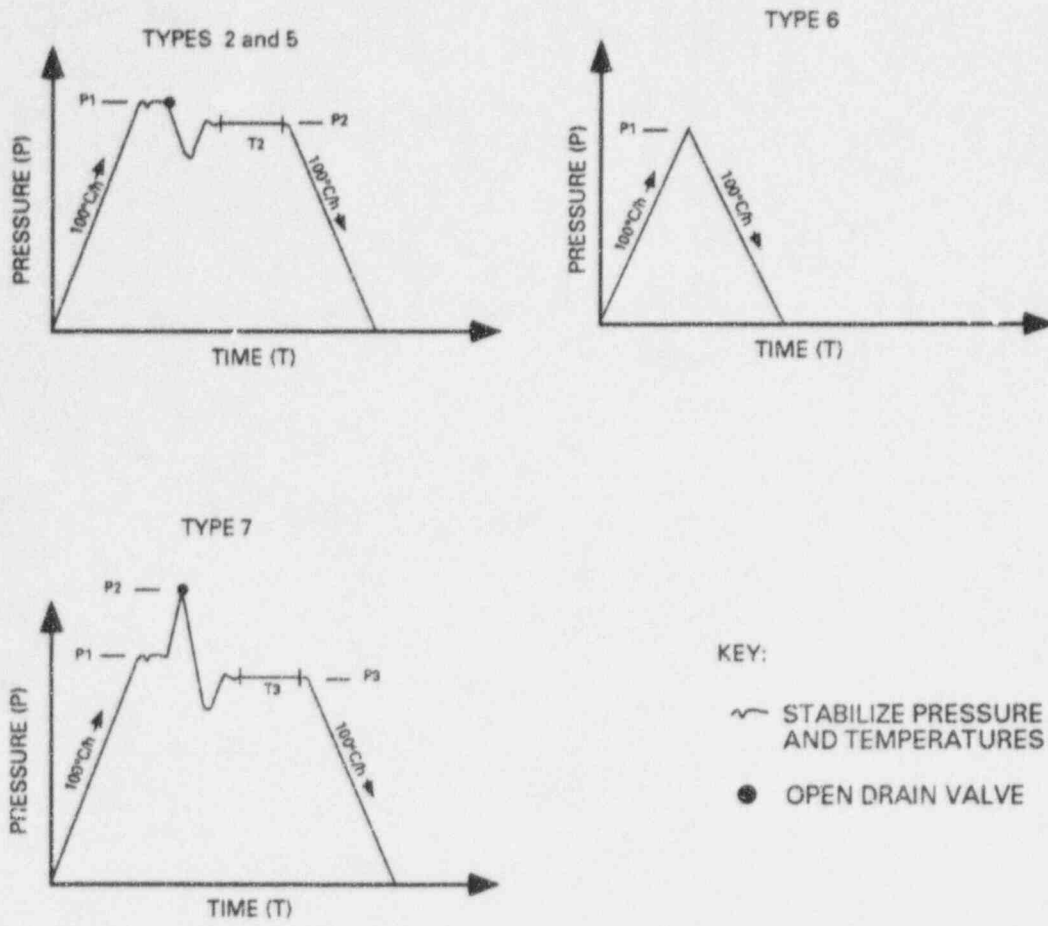


Figure A.3-23 IC Cycle Types

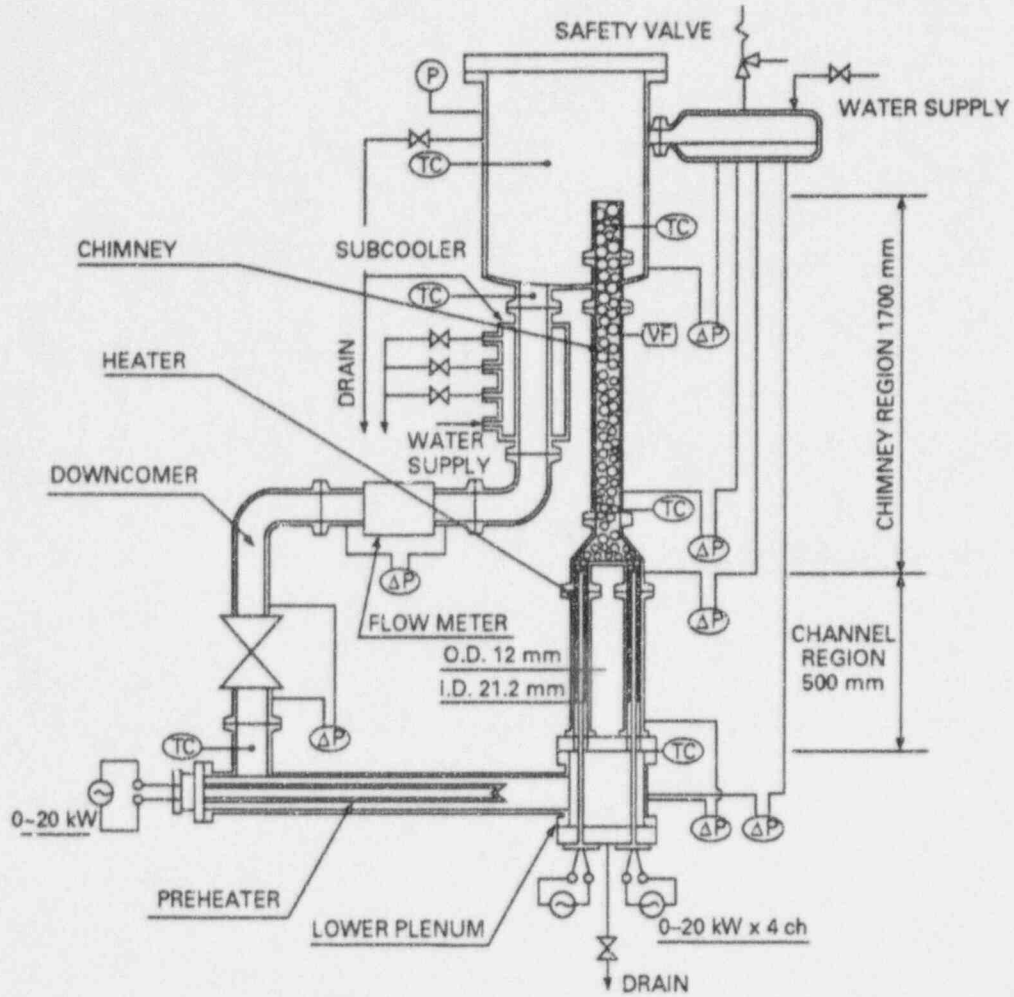


Figure A.3-24 CRIEPI Test Loop Outline

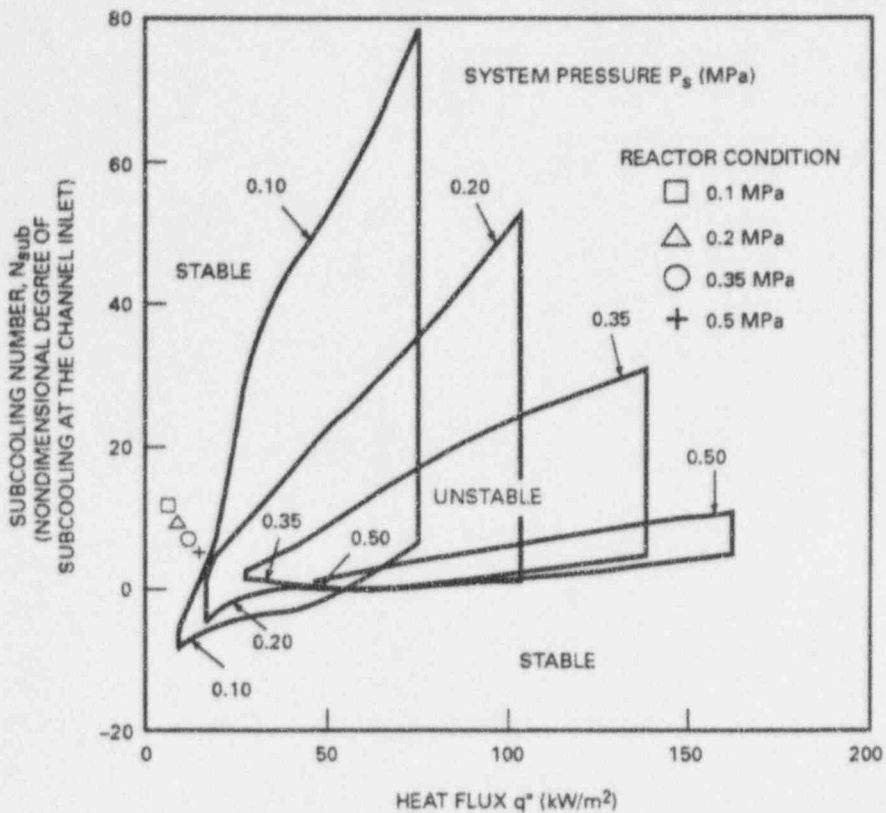


Figure A.3-25 CRIEPI Facility Stability Map Under Lower Pressure Startup Conditions and Representative Parameters for SBWR

ATTACHMENT A1 - TABLE OF CONTENTS FOR TEST AND ANALYSIS DOCUMENTS

Apparent Test Results

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

Data Transmittal Report

1.0 Introduction

- General description and purpose of tests
- Purpose of report

2.0 Objectives

- General Objectives
- Specific Objectives

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

3.0 Test Facility Description

- Detailed description of facility layout
- Scaling study

(Note: Facility descriptions will be from the Test Specification and/or Test Plan and Procedures documents. Scaling is addressed in Reference 32.)

4.0 Instrumentation

- Instrument type and characteristics
- Calibration

5.0 Data Acquisition System

- Hardware configuration
- Data Reduction
- Software

6.0 Test Matrix

- Grouped by type of test

7.0 Test Results

- Grouped by type of test

8.0 Conclusions

- Adequacy of test data
- Applicability to test objectives

9.0 References

Appendices

- A. Instrument List (Type of instrument, number of instrument, measurement, and range)
- B. Modified and Failed Instruments
 - Listed by test
- C. Facility Characterization Tests
 - Pressure drop tests
 - Heat loss tests
- D. Error Analysis
 - Maximum error of measurement
- E. Data Records
 - Format of Data Tapes

Data Analysis Report

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

- 2.0 Objectives
- General Objectives
 - Specific Objectives

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

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

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

- Discussion of observed phenomena

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

- 5.0 References

Preliminary Validation Results

- 1.0 Introduction
 - General description and purpose of tests
 - What tests will be used for assessment
 - How data will be used
- 2.0 Brief description of Test Facility and Test Matrix
 - Referenced to appropriate test reports
- 3.0 Applicability of data to SBWR
 - Range of relevant parameters/ scaling groups compared to SBWR
- 4.0 TRACG model and nodalization
 - Noding used and basis
 - Any modifications for post-test analysis
 - Justification for difference in nodalization vs. SBWR nodalization, if any
 - Discussion of new models, if any
- 5.0 Test Simulation
 - Choice of tests to be simulated
 - Procedure for simulation, including initial and boundary conditions
- 6.0 Qualification results - data vs. predictions
 - Comparisons between data and TRACG results
 - Plots and discussion of key parameters for each test
- 7.0 Results of Assessment
 - Adequacy of TRACG models
 - Implications for SBWR calculations
- 8.0 References

TRACG Qualification For SBWR

Abstract

- 1.0 Introduction
 - 1.1 Relationship to CSAU Process
- 2.0 Qualification Strategy
 - 2.1 Assessment Matrix
 - 2.2 Coverage of PIRT Phenomena
- 3.0 Separate Effects Tests
 - Refer to previous report NEDE-32177P
- 4.0 Component Performance Tests

- 4.1 PANTHERS PCC Performance
- 4.2 PANTHERS IC Performance
- 4.3 PANDA PCC Performance
- 4.4 Suppression Pool Stratification in Blowdown Tests
- 4.5 SLCS Accumulator Performance

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

- 6.0 SBWR Plant Nodalization
 - 6.1 Reactor Vessel
 - 6.2 Containment

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

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

APPENDIX B - SCALING APPLICABILITY

B.1 Scaling Summary

The details of Scaling of SBWR Related Tests is provided in NEDC-32288 [32], which presents a scaling study applicable to the SBWR-related tests. The scope of the study includes:

- (a) a description of the scaling philosophy used for the GIST, GIRAFFE, PANDA, PANTHERS, and single-tube condensation-heat-transfer tests which have been, or will be, conducted in support of the SBWR program;
- (b) the description of a set of scaling laws which are applicable to the SBWR-related test facilities; and
- (c) an evaluation of the test facilities with respect to the proper scaling of the important phenomena and processes identified in the SBWR Phenomena Identification and Ranking Table (PIRT).

The study is fundamentally motivated by the need to demonstrate that the experimental observations from the test programs are representative of SBWR behavior. This includes an identification of any distortions in the representation of the phenomena and the manner in which these distortions can be considered when the experimental data are used for computer code qualification or the development of computer code models.

The Hierarchical Two-Tier Scaling (H2TS) methodology developed by the US Nuclear Regulatory Commission (US NRC) is applied to the extent practical throughout the study. Several scaling considerations addressed by H2TS are automatically satisfied in the SBWR-related experiments where, in all cases, the fluids and their thermodynamic states are prototypical. The various scaling issues are addressed, as appropriate, by either the top-down or bottom-up methodologies embodied in H2TS. The top-down scaling technique, as applied to generic *containment-related processes*, leads to a familiar set of scaling laws with a *system scale* for power, volume, horizontal area in volumes, and mass flow rate, and 1:1 scaling for pressure differences, elevations, and vent submergences.

The scaling of SBWR system components in relation to specific highly-ranked phenomena and processes is conducted according to the bottom-up H2TS methodology. This includes consideration of the following: thermal plumes, mixing and stratification; heat and mass transfer at liquid-gas interfaces; the heat capacity of structures and heat losses; scaling of the vents; and heat and mass transfer in the condensers used for decay heat removal. Finally, the scaling approach followed in designing the various SBWR-related facilities is reviewed in relation to the test objectives. The data collected from these facilities are used in the qualification of the system code TRACG.

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 two-phase level inside the chimney. Outside the vessel, the containment pressure and temperature are used. These studies are an extension of earlier work described in the SSAR which examined the effect of break location on the LOCA and the use of non-ESF systems to prevent core damage.

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

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

C.2 Scenario Definition for Interaction Studies

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

For the several break locations which were analyzed, three conditions on the power availability were considered:

1. Loss of all AC power, except that provided from inverters

The ESF systems, such as GDCS, ADS, and PCCS, are assumed to operate as designed.

2. On-site diesel generator power available

CRD, RWCU and FAPCS are available in addition to the above systems.

3. Normal auxiliary power available

Feedwater system is available in addition to the above systems.

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

C.3 Primary System Interaction Studies

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

C.4 Containment Interaction Studies

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

C.5 Summary of Interaction Studies

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