GE Nuclear Energy

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

Attention: Richard W. Borchardt, Director
Standardization Project Directorate

**Subject: SBWR Technology Reassessment**

The SBWR Technology Reassessment will be held on Tuesday, June 21 through Thursday, June 23, 1994, in the GE office in San Jose, California. The meetings will commence at 8:30 A.M. at 2011 Little Orchard Avenue, Room 1290. Building 2011 is one block north of the GE complex on Little Orchard Avenue, across the street from the BWR Training Facility. Dress is "Business Casual" (no ties), lunch will be available in the GE cafeteria.

The attachments provide background information for the review:

| Attachment | Subject |
|---|---|
| 1 | Preliminary Agenda |
| 2 | Overview of SBWR Systems |
| 3 | Overview of SBWR Performance & Methods |
| 4 | GIST Experiment |
| 5* | GIRAFFE Experiment |
| 6 | PANDA Experiment |
| 7 | PANTHERS Experiment |
| 8* | TRACG Models & Qualification |
| 9 | SBWR Scaling |

**Attachments 5 and 8 above are GE Proprietary and will be provided at the meeting.*

Substantial additional information will be presented at the review.

We look forward to an active and comprehensive review of the technology support to SBWR design certification effort.

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GE has also made arrangements for a group dinner on Tuesday evening, at Fung Lum Restaurant in Campbell. The cost will be $15 per person. If you plan to attend, please notify Pam Pearson on (408) 925-5005.

Any questions may be directed to Bharat Shiralkar on (408) 925-6889 or Terry McIntyre on (408) 925-1441. We look forward to seeing you at the review.

Sincerely,

RHBuchholz for

T. R. McIntyre, Project Manager
SBWR Test Operations & Analysis

Enclosures

cc: M. Malloy, Project Manager (NRC) (w/2 copies of attachments)
F. W. Hasselberg, Project Manager (NRC) (w/1 copy of attachments)

# ATTACHMENT 1

# PRELIMINARY AGENDA

# SBWR TECHNOLOGY REASSESSMENT AGENDA DAY 1

INTRODUCTION - MCINTYRE

AGENDA REVIEW - ALL

SCOPE/OBJECTIVES/ISSUES - SHIRALKAR

REVIEW CHAIRMAN COMMENTS - COOK

BACKGROUND MATERIAL

- SBWR FEATURES AND SYSTEMS - RAO
- SBWR PERFORMANCE AND METHODS - SHIRALKAR
- TRACG MODELS AND QUALIFICATION - ANDERSEN
- SBWR TESTS AND FACILITIES - BILLIG/TORBECK/FITCH

# SBWR TECHNOLOGY REASSESSMENT AGENDA DAY 2

## SCALING

- METHODOLOGY - SHIRALKAR
- TEST FACILITY EVALUATION - GAMBLE

## SBWR UNIQUE FEATURES, ISSUES, AND PHENOMENA

- METHODOLOGY - UPTON
- SUMMARY EVALUATION - SHIRALKAR

## KEY ISSUES AND TECHNOLOGY BASIS

- CHIMNEY ISSUES - SHIRALKAR
- CRITICAL FLOW ISSUES - ANDERSEN
- ISOLATION CONDENSER ISSUES - FITCH
- BORON MIXING ISSUES - ANDERSEN
- FUEL LENGTH ISSUES - SHIRALKAR

# SBWR TECHNOLOGY REASSESSMENT AGENDA DAY 3

## KEY ISSUES AND TECHNOLOGY BASIS (CONTINUED)

- STABILITY - SHIRALKAR
- WETWELL ISSUES - HEALZER
- PASSIVE CONTAINMENT COOLING ISSUES - FITCH

## SYSTEM INTERACTION STUDIES

- SCENARIOS - MARQUINO
- ECCS SCENARIO
  - IC/DPV FOCUS - MARQUINO
  - GDCS/PCCS FOCUS - YANG
- CONTAINMENT SCENARIO
  - PCCS/GDCS FOCUS - HEALZER
  - PCCS/FAPCS FOCUS - CHEUNG

# SBWR TECHNOLOGY REASSESSMENT AGENDA DAY 3 (CONTINUED)

CLOSURE PLAN

- TEST REQUIREMENTS - SHIRALKAR
- TRACG QUALIFICATION PLAN - ANDERSEN

DISCUSSION AND SUMMARY - ALL

# ATTACHMENT 2

# OVERVIEW OF SBWR SYSTEMS

CONTAINS SECTIONS 1.1 AND 1.2 OF THE SBWR SSAR AS REFERENCE MATERIAL
THE FOLLOWING SECTIONS ARE OF PARTICULAR INTEREST TO REVIEWERS:

SECTION 1.2.2.1.2 - NUCLEAR BOILER SYSTEM (INCLUDES AUTOMATIC DEPRESSURIZATION SYSTEM AND DEPRESSURIZATION VALVES)

SECTION 1.2.2.4.2 - ISOLATION CONDENSER SYSTEM

SECTION 1.2.2.4.3 - GRAVITY DRIVEN COOLING SYSTEM

SECTION 1.2.2.6.2 - FUEL AND AUXILIARY POOLS COOLING SYSTEM

SECTION 1.2.2.12 - DC POWER SYSTEM

SECTION 1.2.2.12 - AC POWER SYSTEM

SECTION 1.2.2.14 - CONTAINMENT SYSTEM

# 1.0 Introduction and General Description of Plant

## 1.1 Introduction

### Format and Content

The Simplified Boiling Water Reactor Standard Safety Analysis Report (SBWR SSAR) is written in accordance with Regulatory Guide 1.70. For consistency with NUREG-0800, the SBWR SSAR includes Section 15.8, which addresses anticipated transients without scram, and Chapter 18, which addresses human factors. In addition, treatment of TMI-related matters is presented in Appendix 1A. Failure Modes and Effects Analyses are provided in Appendix 1B, and Compliance with the EPRI Utility Requirements Document (URD) is presented in Appendix 1C.

The response to severe accident policy statement is provided in Chapter 19. Chapter 20 is included to provide a question and response guide.

### SBWR Standard Plant Scope

The SBWR Standard Plant includes all buildings dedicated exclusively or primarily to housing systems and equipment related to the nuclear system or controlled access to these systems and equipment. There are four such buildings within the scope of the SBWR plant. These are:

- reactor building (including containment and main control room);
- turbine building;
- radwaste building; and
- electrical building (including the diesel-generators).

Buildings and structures not in the SBWR Standard Plant scope include the main transformer; switch and; heat sinks for the main condenser, decay heat, and system waste heat; water treatment building; and demineralized water tanks.

In addition to these buildings and their contents, the SBWR Standard Plant provides the supporting facilities shown in Figure 21.1.2-1.

### Engineering Documentation

Engineering documentation for the SBWR Standard Plant is listed on Master Parts List (MPL) No. 18NS07A04. This MPL is a controlled list, structured by system, that contains the identification of hardware and software documentation that defines the SBWR Standard Plant.

### *Type of License Required*

This SBWR SSAR is submitted in support of the application for final design approval (FDA) and design certification (DC) for the SBWR Standard Plant.

### *Number of Plant Units*

For the purpose of this document, only a single standard unit will be considered.

### *Description of Location*

This plant can be constructed at any location which meets the parameters identified in Chapter 2.

### *Type of Nuclear Steam Supply*

This plant has a boiling water reactor nuclear steam supply system designed and supplied by GE and designated as SBWR.

### *Type of Containment*

The SBWR has a low-leakage containment vessel which comprises the drywell and pressure suppression chambe. The containment vessel is a cylindrical steel-lined reinforced concrete structure integrated with the reactor building. The containment nomenclature is specified in Figure 1.1-1.

### *Core Thermal Power Levels*

The information presented herein pertains to one reactor unit with a rated thermal power level of 2000 MWt. The plant uses a direct-cycle, natural circulation boiling water reactor. The reactor system heat balance at rated power is shown in Figure 1.1-2a. The overall plant heat balance is provided on Figures 10.1-2 (guaranteed) and 10.1-3 (valves wide open). The plant operates at a gross electrical power output at rated power of approximately 670 MWe and net electrical power output of approximately 640 MWe.

## 1.1.1 COL License Information

None.

## 1.1.2 References

None.

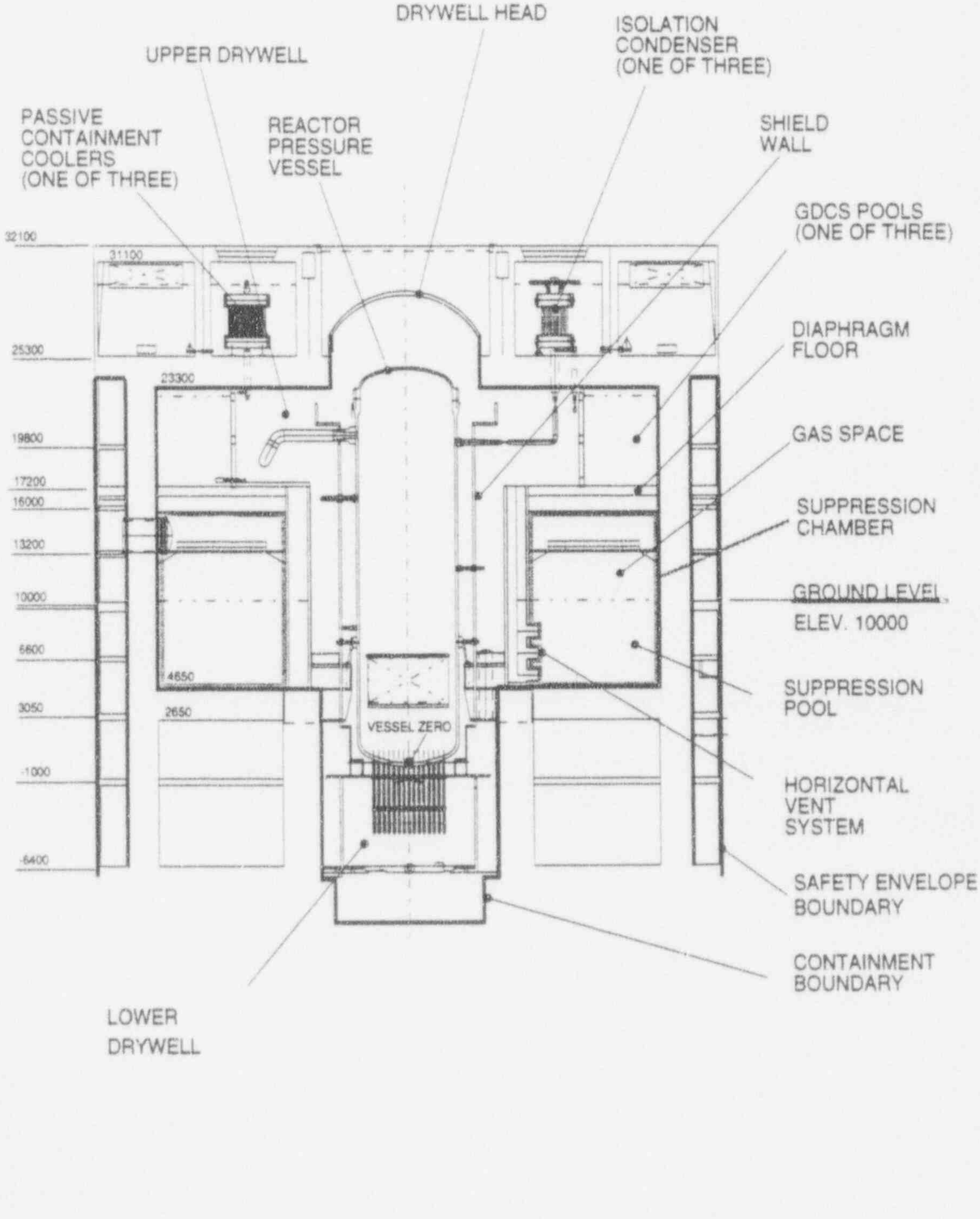



**Figure 1.1-1 Containment Nomenclature**

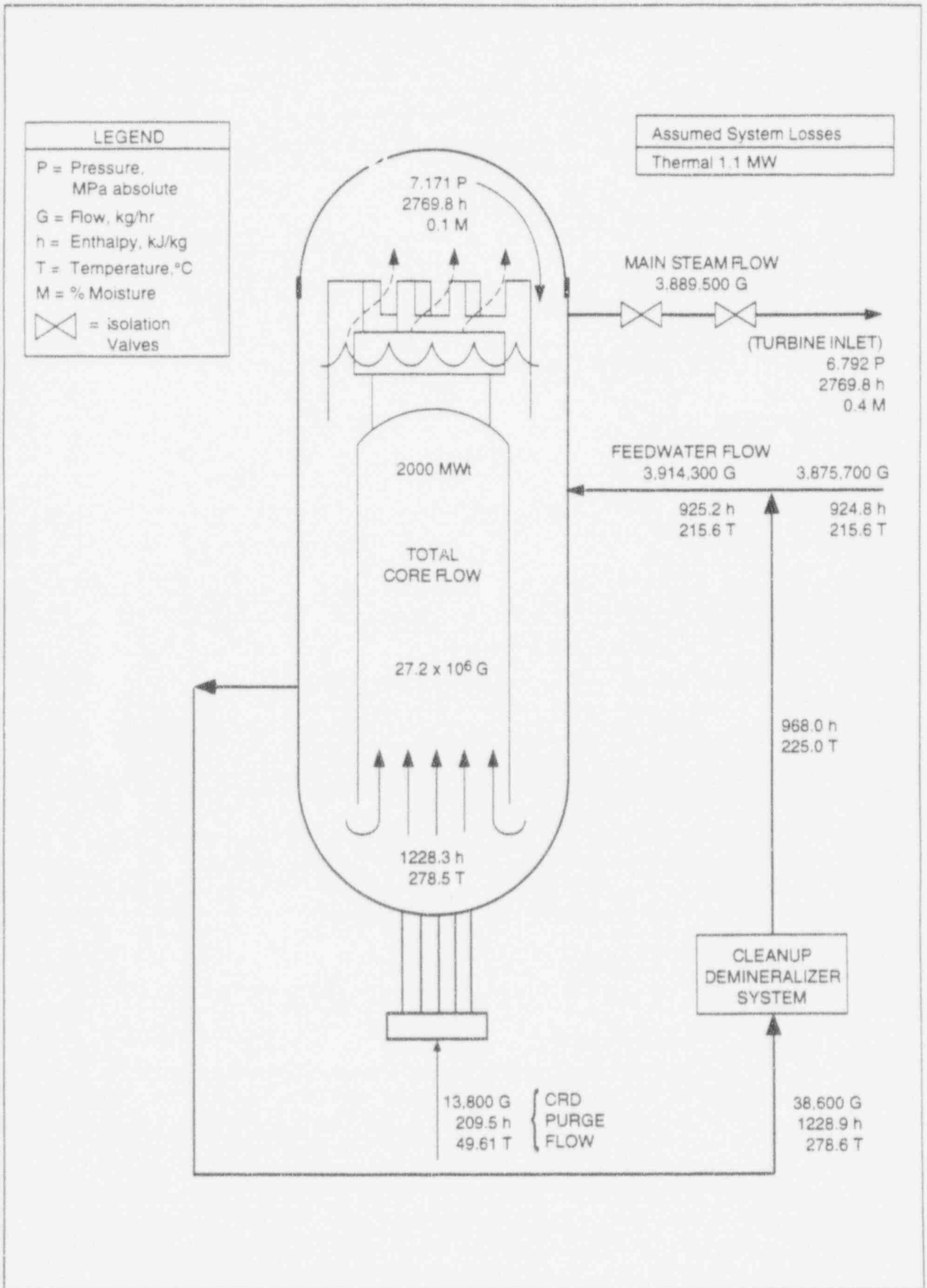



**Figure 1.1-2a Reactor System Heat Balance at 100% Power (SI Units)**

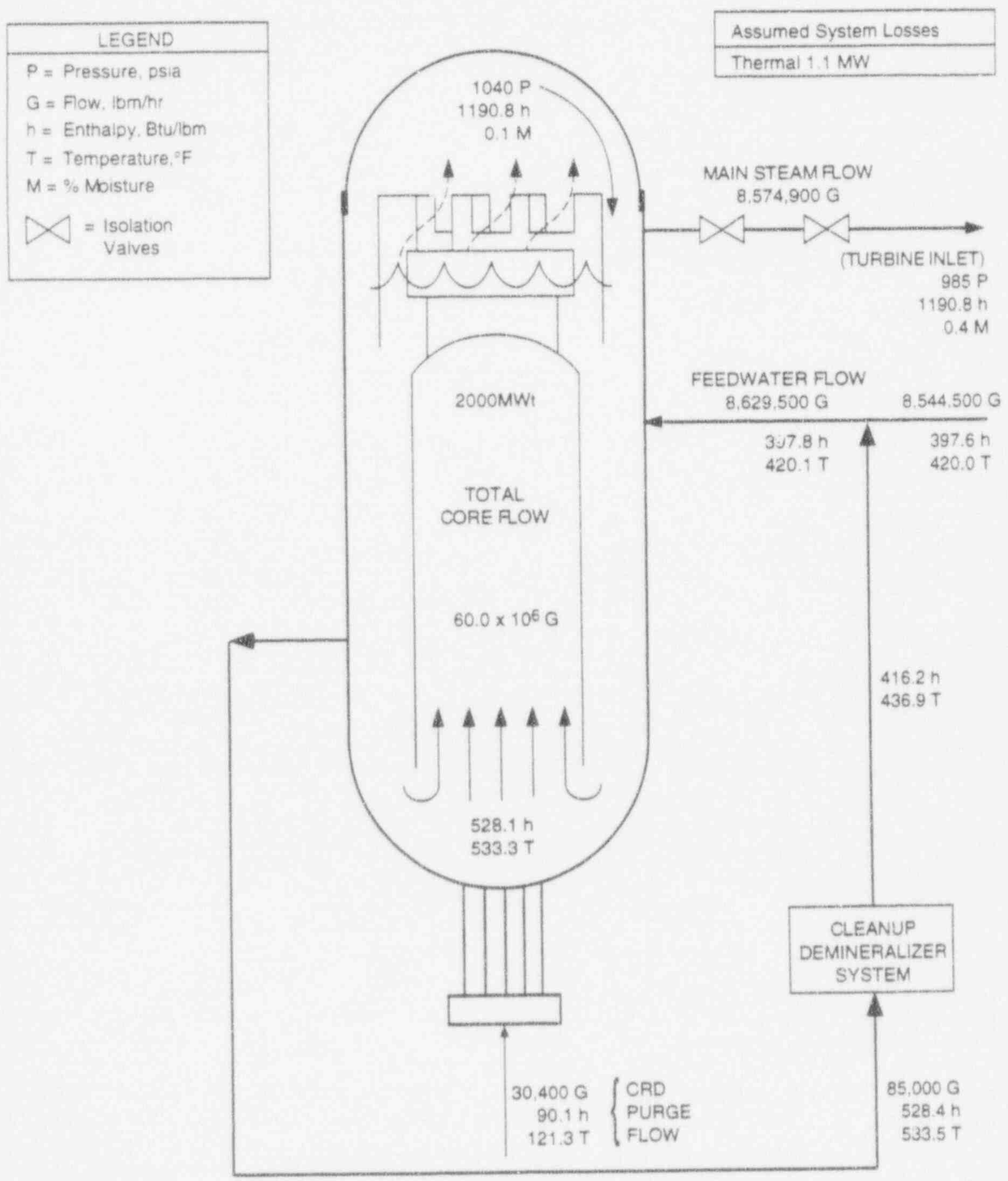



**Figure 1.1-2b Reactor System Heat Balance at 100% Power (English Units)**

# 1.2 General Plant Description

## 1.2.1 Principal Design Criteria

The principal design criteria governing the SBWR Standard Plant are presented in two ways. First, the criteria are classified as applicable to either a power generation function or a safety-related function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear-cut and are sometimes overlapping, the functional classification facilitates safety analysis reviews, while the grouping by system facilitates understanding both the system function and design.

The principal plant structures are shown on Figure 21.1.2-2, sheets 1-20, and are listed below:

- Reactor building — houses all structures, components, equipment and systems providing safety-related functions. This includes the reactor, containment, safety envelope, the refueling area with spent fuel storage, the control room, and auxiliary equipment area.
- Turbine building — houses equipment associated with the main turbine and generator and their auxiliary systems and equipment including the condensate purification system and the process offgas treatment system.
- Radwaste building — houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.
- Electrical building — houses the two non-safety-related standby diesel generators and their associated auxiliary equipment, and the solid-state adjustable speed drive units powering the feedwater pump motors and others powering the Reactor Water Cleanup/Shutdown Cooling System pumps.

### 1.2.1.1 General Power Generation Design Criteria

- The plant is designed to produce electricity from a turbine generator unit using steam generated in the reactor.
- Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.
- Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

- The fuel cladding, in conjunction with other plant systems, is designed to retain integrity so that the consequences of any failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.
- Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.
- Reactor power level is manually controllable.
- Control of the reactor is provided from a single location.
- Reactor controls, including status displays and alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.
- Interlocks or other automatic equipment are provided as backup to procedural control to avoid conditions requiring the functioning of safety-related systems or engineered safety features.
- The station is designed for routine continuous operation whereby steam activation products, fission products, activated corrosion products and coolant dissociation products are processed to remain within acceptable limits.

### 1.2.1.2 General Safety Design Criteria

- The station design conforms to applicable codes and standards as described in Subsection 1.8.2.
- The station is designed, fabricated, erected and operated in such a way that the release of radioactive material to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations, for abnormal transients and for accidents.
- The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.
- The reactor is designed so there is no tendency for divergent oscillation of any operating characteristic considering the interaction of the reactor with other appropriate plant systems.
- The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

- Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered safe by plant analysis.

- Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material containment barrier following abnormal operational transients and to assure cooling of the reactor core following accidents.

- Safety-related systems and engineered safety features function to assure that no damage to the reactor coolant pressure boundary results from internal pressures caused by abnormal operational transients and accidents.

- Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

- Safety-related actions are provided by equipment of sufficient redundancy and independence so that no single failure of active components, or of passive components in certain cases in the long term, will prevent the required actions. For systems or components to which IEEE-279 apply, single failures of either active or passive electrical components are considered in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

- Provisions are made for control of active components of safety-related systems from the control room.

- Safety-related systems are designed to permit demonstration of their functional performance requirements.

- The design of safety-related systems, components and structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.

- Standby electrical power sources have sufficient capacity to power all safety-related systems requiring electrical power concurrently.

- Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

- A containment is provided that completely encloses the reactor systems, drywell, and suppression chambers. The containment employs the pressure suppression concept.

- It is possible to test containment integrity and leak tightness at periodic intervals.

- A safety envelope is provided that basically encloses the containment, with the exception of the areas above the containment top slab and drywell head. The areas above the containment top slab and drywell head are flooded in a pool of water during operation. The safety envelope forms an additional barrier helping to control any potential post-accident containment leakage. The water pool above the containment top slab and drywell head is effective in scrubbing any potential containment leakages through that path.

- The containment and safety envelope in conjunction with other safety-related features limit radiological effects of design basis accidents to less than the prescribed acceptable limits.

- Provisions are made for removing energy from the containment as necessary to maintain the integrity of the containment system following accidents that release energy to the containment.

- Piping that penetrates the containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated when necessary to limit the radiological impact from an uncontrolled release to less than acceptable limits.

- Emergency core cooling is provided to limit fuel cladding temperature to less than the limits of 10CFR50.46 in the event of a design basis loss-of-coolant accident (LOCA).

- The emergency core cooling provides for continuity of core cooling over the complete range of postulated break sizes in the reactor coolant pressure boundary piping.

- Emergency core cooling is initiated automatically when required regardless of the availability of off-site power supplies and the normal generating system of the station.

- The control room is shielded against radiation so that continued occupancy under design basis accident conditions is possible.

- In the event that the control room becomes inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

- Backup reactor shutdown capability independent of normal reactivity control is provided. This backup system has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition.

- Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and off-site dose constraints.

- Systems that have redundant or backup safety-related functions are physically separated, and arranged so that credible events causing damage to one region of the reactor island complex have minimum prospects for compromising the functional capability of the redundant system.

#### 1.2.1.3 Nuclear System Criteria

- The fuel cladding is a radioactive material barrier designed to retain integrity so that failures do not result in dose consequences that exceed acceptable limits through the design power range.

- The fuel cladding in conjunction with other plant systems is designed to retain integrity so that the consequences of any failures are within acceptable limits throughout any abnormal operational transient.

- Those portions of the nuclear system that form part of the reactor coolant pressure boundary are designed to retain integrity as a radioactive material barrier during normal operation and following abnormal operational transients and retain integrity to assure core cooling following accidents.

- The capacity of the heat removal systems provided to remove heat generated in the reactor core for the full range of normal operational transients as well as for abnormal operational transients is adequate to prevent fuel cladding damage that results in dose consequences exceeding acceptable limits.

- The reactor is capable of being shut down automatically in sufficient time to permit decay heat sinks to become effective following loss of operation of normal heat removal systems. The capacity of such systems is adequate to prevent fuel cladding damage.

- The reactor core and reactivity control system are designed such that control rod action is capable of making the core subcritical and maintaining it even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

- Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any operating condition and subsequently to maintain the shutdown condition.

- The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

### 1.2.1.4 Electrical Power Systems Criteria

Sufficient normal, auxiliary, and standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required safety -related functions under all postulated accident conditions.

### 1.2.1.5 Auxiliary Systems Criteria

- Other auxiliary systems, such as service water, cooling water, fire protection, heating and ventilating, communications, and lighting, are designed to function as needed during normal and/or accident conditions.

- Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe condition are designed so that a failure of these systems shall not prevent the safety-related systems from performing their design functions.

### 1.2.1.6 Shielding and Access Control Criteria

Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any normal mode of plant operation.

### 1.2.1.7 Power Conversion Systems Criteria

Components of the power conversion systems are designed to attain the following basic objectives:

- The components of the power conversion systems are designed to produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater with a major portion of its gases and particulate impurities removed.

- The components of the power conversion systems are designed so that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

### 1.2.1.8 Nuclear System Process Control Criteria

- Control equipment is provided to allow the reactor to respond automatically to load changes within design limits.

- Manual control of the reactor power level is provided.

- Nuclear systems process displays, controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

#### 1.2.1.9 Electrical Power System Process Control Criteria

- The Class 1E power systems are designed with four divisions with any two divisions being adequate to safely place the unit in the safe shutdown condition.

- Protective relaying is used, in the event of equipment failure, to detect and isolate faulted equipment from the system with a minimum of disturbance to uninvolved systems or equipment.

- Two standby diesel generators are started and connected to both safety-related and non-safety-related loads if both the preferred and alternate power sources are lost. If these non-Class 1E DGs are also inoperable, all safety-related loads will be powered by the Class 1E divisional batteries.

- Safety-related electrical systems and components are monitored in the control room.

## 1.2.2 Plant Description

### 1.2.2.1 Nuclear Steam Supply

#### 1.2.2.1.1 Reactor Pressure Vessel

The reactor pressure vessel (RPV) assembly consists of the pressure vessel and its appurtenances, supports and insulation, and the reactor internals enclosed by the vessel (excluding the core, in-core nuclear instrumentation, neutron sources, control rods, and control rod drives).

The reactor coolant pressure boundary (RCPB) of the RPV retains integrity as a radioactive material barrier during normal operation and following abnormal operational transients and retains integrity to contain coolant during design basis accidents (DBAs).

Certain RPV internals support the core and support instrumentation used during a DBA. Other RPV internals direct coolant flow, separate steam from the steam/water mixture leaving the core, hold material surveillance specimens, and support instrumentation used for normal operation.

The RPV, together with its internals, provides guidance and support for the fine-motion control rod drives (FMCRDs). Certain of the reactor internals distribute sodium pentaborate solution delivered by the Standby Liquid Control (SLC) System when necessary to achieve core subcriticality via means other than inserting of control rods.

The RPV restrains the FMCRDs to prevent ejection of a control rod connected with a drive in the event of a postulated failure of a drive housing.

***Reactor Pressure Vessel***

The RPV consists of a vertical, cylindrical pressure vessel of welded construction, with a removable top head, and head flanges, seals and bolting. The vessel also includes penetrations, nozzles, shroud support, and venturi shaped flow restrictors in the steam outlet nozzles. The shroud support carries the weight of peripheral fuel assemblies, neutron sources, core plate, top guide, chimney shroud and chimney head with steam separators and dryers, and it laterally supports the fuel assemblies. An integral reactor vessel skirt supports and anchors the vessel on the RPV support structure in the containment.

The reactor vessel is 6 meters (236 in.) in diameter minimum, with a wall thickness of about 158 mm (6.2 in.)with cladding, and 24.5 m (80.4 ft.) tall from the inside of the bottom head (elevation zero) to the inside of the top head. The bottom of the active fuel location is 3750 mm (147.6 in.) from elevation zero and the active core is 2743 mm (108 in.) high.

The overall RPV height of approximately 25 m (82 feet) permits natural circulation driving forces to produce abundant core coolant flow. An increased internal flow-path length relative to prior BWRs is provided by a long "chimney" in the space which extends from the top of the core to the entrance to the steam separator assembly. The chimney and steam separator assembly are supported by a shroud assembly which extends to the top of the core. The large RPV volume provides a large reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery can occur as a result of feedwater flow interruption or a LOCA. This gives an extended period of time during which automatic systems or plant operators can reestablish reactor inventory control using any of several normal, non-safety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety equipment. The large RPV volume also reduces the reactor pressurization rates that develop when the reactor is suddenly isolated from the normal heat sink which eventually leads to actuation of the safety-relief valves.

The FMCRDs are mounted into permanently attached CRD housings. The CRD housings extend through, and are welded to CRD penetrations (stub tubes) formed in the RPV bottom head.

A flanged nozzle is provided in the top head for bolting on of the flange associated with the instrumentation for the initial vibration test of internals.

An integral reactor vessel skirt supports the vessel on the RPV support structure. Steel anchor bolts extend through a steel structure comprising the upper part of the RPV support structure, securing the flange of the skirt to the structure. Stabilizers help the upper portion of the RPV resist horizontal loads. Lateral support among the CRD housings and in-core housings are provided by restraints which, at the periphery, are supported from CRD housing restraint beams.

The RPV insulation is supported from the shield wall surrounding the vessel. Insulation for the upper head and flange is supported by a steel frame independent of the vessel and piping. Insulation access panels and insulation around penetrations are designed for ease of installation and removal for vessel Inservice Inspection and maintenance operations.

The RCPB portions of the RPV and appurtenances are classified as Quality Group A, Seismic Category I. The design, materials, manufacturing, fabrication, testing, examination, and inspection used in the construction meet the requirements of ASME Code, Section III, Subsection NB, Class 1 Components. The RPV support skirt, stabilizers, CRD housing restraints and in-core housing restraints are Seismic Category I and are designed and constructed to meet the requirements of ASME Code, Section III, Subsection NF, Component Supports. The shroud support is classified as Seismic Category I, and designed and fabricated to meet the requirements of ASME Code Class CS (core support structures). Hydrostatic tests of the RPV are performed in accordance with the requirements for ASME Code Class 1 vessels. The components are code-stamped according to their code class.

The RPV materials comply with the provisions of the ASME Code Section III, Appendix I, Subsection NB-2000, and meet the specification requirements of 10CFR50, Appendix G.

The RPV is constructed primarily from low alloy, high strength steel plate and forgings. Plates are ordered to ASME SA-533, TYPE B, Class 1, and forgings to ASME SA-508, Class 3. These materials are melted to fine grain structure and are supplied in the quenched and tempered condition. Further restrictions include a requirement for vacuum degassing to lower the hydrogen level and improve the cleanliness of the low alloy steels. Materials used in the core beltline region also specify limits of 0.05% maximum copper and 0.012% maximum phosphorous content in the base materials and a 0.08% maximum copper and 0.012% maximum phosphorous content in weld materials. The maximum sulfur content for the base material and weld material is 0.01%.

Studs, nuts, and washers for the head flange are ordered to ASME SA-540, Grade B23 or GRADE B24 having minimum yield strength level of 893 MPa (129,500 psi). The maximum measured ultimate tensile strength of the stud bolting materials do not exceed 1172 MPa (170,000 psi).

Electroslag welding is not applied for structural welds. Preheat and interpass temperature employed for welding of low alloy steel meet or exceed the values given in ASME, Section III, Appendix D. Post-weld heat treatment at 593°C (1099°F) minimum and not exceeding 635°C (1175°F) is applied to all low-alloy steel welds. Welding electrodes for low alloy steel are low hydrogen type ordered to ASME SFA-5.5.

Pressure boundary welds are given an ultrasonic examination in addition to the radiographic examination performed during fabrication. The ultrasonic examination method, including calibration, instrumentation, scanning sensitivity, and coverage, is based on the requirements imposed by ASME, Section XI, Appendix I. Acceptance standards are equivalent or more restrictive than required by ASME, Section XI.

A stainless steel weld overlay is applied to the interior of the cylindrical shell and the steam outlet nozzle. Other nozzles and the top head do not have cladding. The bottom head is clad with Ni-Cr-Fe alloy.

Fracture toughness tests of pressure boundary ferritic materials, weld metal and heat-affected zone (HAZ) materials are performed in accordance with the requirements for ASME Code Class 1 vessels. Both longitudinal and transverse specimens are used to determine the minimum upper shelf energy (USE) level of the core beltline materials. Separate, unirradiated baseline specimens are used to determine the transition temperature curve of the core beltline base materials, weld metal, and HAZ materials.

For the vessel material surveillance program, specimens are manufactured from the material actually used in the reactor beltline region and welds typical of those in the beltline region, thus representing base metal, weld material, and the HAZ material. The plate and weld specimens are heat treated in a manner which simulates the actual heat treatment performed on the core region shell plates of the completed vessel. Each in-reactor surveillance capsule contains Charpy V-notch specimens of base metal, weld metal, and HAZ material, and tensile specimens from base metal and weld metal. Brackets are welded to the vessel cladding in the core belt region for retention of the detachable holders, each of which contains a number of the specimen capsules. Neutron dosimeters and temperature monitors are located within the capsules.

Access for examinations of the installed RPV is incorporated into the design of the vessel, reactor shield wall, and vessel insulation.

#### *Reactor Internals*

The reactor internals consist of core support structures and other equipment.

The core support structures locate and support the fuel assemblies, form partitions within the reactor vessel to sustain pressure differentials across the partitions, and direct the flow of coolant water. The structures consists of a shroud, shroud support, coreplate, top guide, and integral fuel support and control rod guide tubes (CRGTs).

The other reactor internals consist of control rods, feedwater spargers, SLC distribution headers, in-core guide tubes, surveillance specimen holders, chimney, chimney partitions, chimney head, steam separator assembly, and the steam dryer assembly.

The shroud support, shroud, and chimney make up a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downward recirculation flow outside the core. This partition separates the core region from the downcomer annulus.

The core plate consists of a circular stainless steel plate with round openings and is stiffened with a beam structure. The core plate provides lateral support and guidance for the integral support and CRTGs, in-core flux monitor guide tubes, peripheral fuel supports and startup neutron sources. The last two items are also supported vertically by the core plate.

The top guide consists of a circular plate with square openings for fuel and with a cylindrical side forming an upper shroud extension. Each opening provides lateral support and guidance for four fuel assemblies or, in the case of peripheral fuel, less than four fuel assemblies. Holes are provided in the bottom surface of the top guide where the sides of the openings intersect, to anchor the in-core instrumentation detectors and start-up neutron sources.

The fuel assemblies are vertically supported in two ways depending upon whether they are located next to a control rod or not. The peripheral fuel assemblies which are located at the outer edge of the active core, not adjacent to a control rod, are supported by the peripheral fuel supports. The peripheral fuel supports are welded to the core plate and each support one assembly. The peripheral fuel supports contain flow restricting sections to provide coolant flow to the fuel assembly. The remaining fuel assemblies which are adjacent to the control rods are supported by the integral fuel support and CRGTs. Each integral fuel support and CRGT supports four fuel assemblies vertically upward and provides lateral support to the bottom of the fuel. The fuel support forms the top part of the integral unit with the bottom section forming the CRGT. The integral fuel support and guide tube are laterally supported by the core plate.

The CRGTs section is cruciform in shape and is designed as a guide for the lower end of the control rod. The lower end of the CRGT section is supported by the control rod drive (CRD) housing, which in turn transmits the weight of the integral fuel support and CRGT, and the four fuel assemblies to the reactor vessel bottom head. The lower end of the CRD housing is welded to a stub tube which is directly welded to the bottom of the vessel. Coolant flow which has entered the lower plenum of the vessel travels upward, adjacent to the guide tube section and enters the fuel support section just below the core plate. The fuel support section contains four flow restricting openings which control coolant flow to the fuel assemblies.

The base of the CRGT section is provided with a device for coupling to the FMCRD. The CRD is restrained from ejection, in the case of a stub tube to CRD housing weld failure, by the coupling of the drive with the guide tube base. In this event, the fuel support flange will contact the core plate and thus restrain the ejection. The coupling will also prevent ejection if the CRD housing fails below the stub tube weld. In this event, the integral guide tube and fuel support remains supported by the CRD housing left intact above the stub tube weld.

The control rods are cruciform-shaped neutron absorbing members that can be inserted or withdrawn from the core by the FMCRD to control reactivity and reactor power.

Each of the four feedwater lines is connected to four spargers via four RPV nozzles. The feedwater spargers are stainless steel headers located in the mixing plenum above the downcomer annulus. Each sparger, in two halves, with a tee connection at the middle, is fitted to the corresponding RPV feedwater nozzle. The sparger tee inlet is connected to the RPV nozzle safe end by a double thermal sleeve arrangement. Feedwater flow enters the center of the spargers and is discharged radially inward to mix the cooler feedwater with the downcomer flow from the steam separators and steam dryers.

In-core guide tubes (ICGTs) protect the in-core flux monitoring instrumentation from flow of water in the bottom head plenum. The ICGTs extend from the top of the in-core housing to the top of the core plate. The local power range monitoring (LPRM) detectors for the power range neutron monitoring (PRNM) subsystem and the detectors for the startup range neutron monitoring (SRNM) subsystem are inserted through the guide tubes.

Two levels of stainless steel stabilizer latticework of clamps, tie bars, and spacers give lateral support and rigidity to the ICGTs. The stabilizers are connected to the shroud and shroud support.

Surveillance specimen capsules, which are held in capsule holders mentioned earlier, are located at three azimuths at a common elevation in the core beltline region. The capsule holders are non-safety-related internals. The capsule holders are mechanically retained by capsule holder brackets welded to the vessel cladding that allow capsule removal and re-installation.

As a natural circulation reactor the SBWR requires additional elevation head created by the density difference between the saturated water-steam mixture exiting the core and the subcooled water exiting the region just below the separators and the feedwater inlet. The chimney provides this elevation head or driving head necessary to sustain the natural circulation flow. The chimney is a long cylinder mounted to the top guide and which supports the steam separator assembly. The chimney forms the annulus separating the subcooled recirculation flow returning downward from the steam separators and feedwater. from the upward steam-water mixture flow exiting the core. Inside the chimney are partitions which separate groups of 36 fuel assemblies and thereby form smaller chimney sections limiting cross flow and flow instabilities.

The BWR direct cycle requires separation of steam from the steam-water mixture leaving the core. This is accomplished inside the RPV by passing the mixture sequentially first through an array of steam separators attached to a removable cover on the top of the chimney assembly, and then through standard BWR steam dryers. The steam dryer and the separator assembly are connected so they can be removed as a unit assembly to simplify refueling. The dryer arrangement has been sized to dry steam with an inlet moisture content of 20%, although inlet moisture of less than 2% is expected during normal operations. The dryers are designed to provide outlet dry steam with a moisture content ≤ 0.1%

The core support structures are classified as Quality Group C, Seismic Category I. The design, materials, manufacturing, fabrication, examination, and inspection used in the construction of the core support structures meet the requirements of ASME Code Section III, subsection NG, Core Support Structures.

These structures are code-stamped accordingly. Other reactor internals are designed per the guidelines of ASME Code NG-3000 and are constructed so as not to adversely affect the integrity of the core support structures as required by NG-1122.

Special controls on material fabrication processes are exercised when austenitic stainless steel is used for construction of RPV internals in order to avoid stress corrosion cracking during service.

Design and construction of the RPV internals assure that the internals can withstand the effects of flow-induced vibration (FIV).

#### 1.2.2.1.2 Nuclear Boiler System

The primary functions of the Nuclear Boiler System (NBS) are: (1) to deliver steam from the RPV to the turbine main steam system (TMSS), (2) to deliver feedwater from the condensate and feedwater system (C & FS) to the RPV, (3) to provide overpressure protection of the RCPB, (4) to provide automatic depressurization of the RPV in the event of a LOCA where the RPV does not depressurize rapidly, and (5) with the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV such as RPV pressure, metal temperature, and water level instrumentation.

The main steam lines (MSLs) are designed to direct steam from the RPV to the TMSS; the feedwater lines (FWLs) to direct feedwater from the C & FS to the RPV; the RPV instrumentation to monitor the conditions within the RPV over the full range of reactor power operation.

The NBS contains the valves necessary for isolation of the MSLs, FWLs, and their drain lines at the containment boundary.

The NBS contains the safety-relief valve discharge lines (SRVDLs) including the steam quencher located in the suppression pool at the end of each SRVDL.

The NBS also contains the RPV head vent line and non-condensable gas removal line.

### Main Steam Lines

The NBS contains the portion of the MSLs from their connection to the RPV to the boundary with the TMSS which occurs at the seismic interface located downstream of the outboard main steam isolation valves (MSIVs).

The main steam lines are Quality Group A from the RPV out to and including the outboard MSIVs, and Quality Group B from the outboard MSIVs to the turbine stop valves. They are Seismic Category I from the RPV out to the seismic interface.

### Main Steamline Flow Limiter

The main steam line flow limiter is essentially a flow restricting venturi built into the RPV MSL nozzle of each of the two main steam lines. The restrictor limits the coolant blowdown rate from the reactor vessel to a (choke) flow rate equal to or less than 200% of rated steam flow at 7.07 MPa (1025 psig) upstream gauge pressure in the event a main steam-line break occurs anywhere downstream of the nozzle. The MSL flow limiters thus limit off-site dose from postulated MSL breaks outside containment, while the MSIVs are closing. They limit the 2-phase depressurization level swell and liquid coolant loss from the vessel, and the rate of first-peak (vent clearing) containment pressure rise for a MSL break inside containment. The flow limiters also limit the intensity of the depressurization level swell and differential pressures momentarily developed on core internals following a MSL break.

The flow restrictors are designed and fabricated in accordance with the ASME Code and designed in accordance with ASME Fluid Meters Handbook. The flow restrictor has no moving parts.

The restrictors are also used to monitor steam flow and to initiate closure of the main steamline isolation valves when the steam flow exceeds preselected operational limits. The vessel dome pressure and the venturi throat pressure are used as the high and low pressure sensing locations.

### Main Steam Isolation Valves

Each main steam isolation valve assembly consists of a main steam isolation valve (MSIV), a pneumatic accumulator, connecting piping and associated controls.

There are two MSIVs welded into each of the two MSLs. On each MSL there is one MSIV in the containment and one MSIV outside the containment. Each set of two MSIVs isolate their respective MSL upon receipt of isolation signal and will close on loss of pneumatic pressure to the valve.

The MSIVs are Y-pattern globe valves. The main disc or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed. The Y-pattern configuration permits the inlet and outlet flow passages to be streamlined; this minimizes pressure drop during normal steam flow.

The primary actuation mechanism uses a pneumatic cylinder; the speed at which the valve opens and closes can be adjusted. Helical springs around the spring guide shafts will close the valve if gas pressure in the actuating cylinder is reduced.

The MSIV quick-closing speed is ≥ 3 and ≤ 4.5 seconds when $N_2$ or air pressure is admitted to the upper piston compartment. The valve can be test closed with a 45-60 second slow closing speed by admitting $N_2$ or air to both the upper and lower piston compartments.

### Feedwater Lines

The feedwater piping consists of two FWLs connecting to a feedwater supply header. Isolation of each FWL is accomplished by two containment isolation valves consisting of one check valve inside the drywell and one positive closing check valve outside the containment. Also included in this portion of the FWL is a manual maintenance valve located between the inboard isolation valve and the reactor nozzle. The feedwater line upstream of the outboard isolation valve contains an additional check valve, a remote manual motor-operated (MO) gate valve, and a seismic interface restraint. The outboard isolation valve and the MO gate valve provide a quality group transitional point in the FWLs.

The feedwater piping is Quality Group A from the RPV out to and including the outboard isolation valve, Quality Group B from the outboard isolation valve to and including the MO gate valve, and Quality Group D upstream of the MO gate valve. The feedwater piping and all connected piping 2 1/2-inch or larger nominal size are Seismic Category I from the RPV to the seismic interface.

### Safety/Relief Valves

The nuclear pressure relief system consists of safety/relief valves (SRVs) located on the MSLs between the RPV and the inboard MSIV. There are four SRVs per MSL. SRVs provide three main protection functions:

(1) Overpressure safety operation: The valves function as safety valves and open to prevent nuclear system overpressurization. They are self-actuating by inlet

steam pressure.

The safety mode of operation is initiated when direct and increasing static inlet steam pressure overcomes the restraining spring and frictional forces acting against the inlet steam pressure at the valve disc. This then moves the disc in the opening direction. The condition at which this actuation is initiated corresponds to the set-pressure value stamped on the nameplate of the SRV.

(2) Overpressure relief operation: The SRVs can be operated individually in the power-actuated mode by remote manual switches located in the main control room (MCR). The valves are opened using a pneumatic actuator to reduce pressure or to limit pressure rise.

This mode of operation is initiated when an electrical signal is received at any of the solenoid valves located on the pneumatic actuator assembly. The solenoid valve(s) open, allowing pressurized air to enter the lower side of the pneumatic cylinder which pushes the piston and rod upwards. This action pulls the valve disc lifting mechanism to allow steam to discharge through the SRV. When the solenoids are deenergized, the piston and rod fall downward which causes the valve to reseat and stop SRV steam flow.

The SRV pneumatic operator is so arranged that, if it malfunctions, it will not prevent the SRV from opening when steam inlet pressure reaches the spring lift setpoint.

(3) Depressurization operation: This is discussed separately, below.

The SRVs meet the requirements of ASME Code Section III. The power supply is 125 volts dc, Class 1E for the system. The SRV controls are classified as Class 1E.

Each SRV has one dedicated, independent pneumatic accumulator which provides the safety-related, assured nitrogen supply for opening the valve.

The SRVs are flange mounted onto forged outlet fittings located on the top of the main steamline piping in the drywell. The SRVs discharge through lines routed to quenchers in the suppression pool.

### *Automatic Depressurization Subsystem*

The Automatic Depressurization Subsystem (ADS) quickly depressurizes the RPV in sufficient time for the Gravity-Driven Cooling System (GDCS) injection flow to replenish core coolant to maintain core temperature below design limits in the event of a LOCA. It also maintains the reactor depressurized for continued operation of GDCS after an accident without need for power.

The ADS consists of the eight SRVs and six depressurization valves (DPVs) and their associated instrumentation and controls.

Four DPVs are flange-mounted on horizontal stub lines connected to the RPV at about the elevation of the MSLs. The other two DPVs are flange-mounted on horizontal lines branching from each MSL. The DPVs discharge into the drywell.

The SRVs and DPVs are actuated in groups of valves at staggered times as the reactor undergoes a relatively slow depressurization. This minimizes reactor level swell during the depressurization, thereby enhancing the passive resupply of coolant by the GDCS. The staggered opening of the valves is achieved by delay timers.

The use of a combination of SRVs and DPVs to accomplish the ADS function provides an improvement in ADS reliability against hypothetical common-mode failures of otherwise non-diverse ADS components. It also minimizes components and maintenance as compared to using only SRVs or only DPVs for this function. By using the SRVs for two different purposes, the number of DPVs required is minimized. By using DPVs, which have about twice the steam relieving capacity of the SRVs, for the additional depressurization capability needed beyond what the SRVs can provide, the total number of SRVs, SRV discharge lines, and quenchers in the suppression pool is minimized. The need for SRV maintenance, periodic calibration and testing, and the potential for simmering are minimized with this arrangement.

The ADS automatically actuates on a reactor Level 1 signal that persists for a preset time. A two-out-of-four Level 1 logic is used to activate the SRVs and DPVs. The persistence requirement for the Level 1 signal ensures that momentary system perturbations do not actuate ADS when it is not required. The two-out-of-four logic assures that a single failure will not cause spurious system actuation while also assuring that a single failure cannot prevent initiation. The ADS may also be manually initiated from the main control room.

***Depressurization Valves***

The DPVs are of a non-leak/non-simmer/non-maintenance design. They are straight-through, squib-actuated, non-reclosing valves with a metal diaphragm seal. The valves are connected to an 8 in. inlet pipe and a 12 in. outlet pipe. Each valve provides about twice the depressurization capacity as an SRV. The DPV is closed with a cap covering the inlet chamber. The cap will readily shear off when pushed by a valve plunger which is actuated by the explosive initiator-booster. This opens the inlet hole through the plug. The sheared cap is hinged such that it drops out of the flow path and will not block the valve. The DPVs are designed so that there is no leakage across the cap throughout the life of the valve.

Two initiator-boosters (squibs) actuate the shearing plunger, which are in turn initiated by any one of, or any combination of, three battery-powered, independent firing circuits. One initiator-booster has two pairs of pins connected through a wire bridge, the other has one pair of pins connected through a bridge wire. The firing of one initiator-booster is adequate to activate the plunger. Nominal firing voltage is 125 volts dc, however the initiator-boosters are designed to function with any applied voltage between 90 and 155 volts dc. The valve design and initiator-booster design is such that there is substantial thermal margin between operating temperature and the self-ignition point of the initiator-booster.

The DPVs form a part of the reactor coolant pressure boundary (RPCB) and are therefore Quality Group A, ASME Section III, Class 1, and Seismic Category I.

***NBS Instrumentation***

The NBS RPV instrumentation monitors and provides control inputs for operational variables during plant operation.

The NBS contains the instrumentation for monitoring the reactor pressure, metal temperature, and water level. The reactor pressure and water level instruments are used by multiple systems, both safety-related and non-safety-related.

Pressure indicators and transmitters detect reactor vessel internal pressure from the same instrument lines used for measuring reactor vessel water level.

RPV coolant temperatures are determined by measuring saturation pressure (which gives the saturation temperature), outlet flow temperature to the RWCU/SDC System, and RPV bottom head drain line temperature. Reactor vessel outside surface (metal) temperatures are measured at the head flange and the bottom head locations. Temperatures needed for operation and for operating limits are obtained from these measurements. During normal operation, either reactor steam saturation temperature and/or inlet temperatures of the reactor coolant to the RWCU/SDC system and the RPV bottom head drain can be used to determine the RPV coolant temperature.

The instruments that sense the water level are differential pressure devices calibrated for a specific RPV pressure (and corresponding liquid temperature). The water level measurement instrumentation is the condensate reference chamber type. Instrument reference zero for all the RPV water level ranges is the top of the active fuel. The following is a description of each water level range.

(1) Shutdown Range Water Level

This range is used to monitor the reactor water level during shutdown conditions when the reactor system is flooded for maintenance and head removal. The two RPV instrument taps used for this water level measurement are located at the top of the RFV head, and just below the dryer skirt.

(2) Narrow Range Water Level

This range is used to monitor reactor water level during normal power operation. This range uses the RPV taps near the top of the steam outlet nozzles and near the bottom of the dryer skirt. The Feedwater Control (FDWC) System uses this range for its water level control and indication inputs. The RPS also uses this range for scram initiation.

(3) Wide Range Water Level

This range is used to monitor reactor water level for events where the water level exceeds the range of the narrow range water level instrumentation, and is used to generate the low reactor water level trip signals which indicate a potential LOCA. This range uses the RPV taps at the elevations near the top of the steam outlet nozzles and the nearest tap above the top guide.

(4) Fuel Zone Range Water Level

This range is provided for post-accident monitoring and provides the capability to monitor the reactor water level below the wide range water level instrumentation. This range uses the RPV taps at the elevations near the top of the steam outlet nozzles and the taps below the bottom of active fuel.

Thermocouples are located in the discharge exhaust pipes of the SRVs. The temperature signals go to a multipoint recorder with an alarm and will be activated by any temperature in excess of a set temperature, signaling that one of the SRV seats has started to leak.

Control room indication and alarms are provided for the important plant parameters monitored by the NBS.

***NBS ASME Code Requirements***

The major NBS mechanical components are designed to meet ASME Code Requirements as shown below:

| Component | ASME Code Class | Design Conditions | |
|---|---|---|---|
| | | Gauge Pressure | Temperature |
| FWLs from the MOVs to the outboard containment isolation check valves | 2 | 8.62 MPa (1250 psig) | 302°C (575°F) |
| FWLs from the outboard containment isolation check valve to the RPV | 1 | 8.62 MPa (1250 psig) | 302°C (575°F) |
| FWL line outboard containment isolation check valve | 1 | 8.62 MPa (1250 psig) | 302°C (575°F) |
| Main steam isolation valves (MSIVs) | 1 | 9.48 MPa (1375 psig) | 308°C (586°F) |
| Safety/relief valves (SRVs) | 1 | 9.48 MPa (1375 psig) | 308°C (586°F) |
| Main steam lines (MSLs), from RPV to outboard MSIVs | 1 | 8.62 MPa (1250 psig) | 302°C (575°F) |
| MSLs from the outboard MSIVs to the seismic interface restraint | 2 | 8.62 MPa (1250 psig) | 302°C (575°F) |
| SRV discharge line piping, from the SRVs to the vent wall penetration | 3 | 3.72 MPa (540 psig) | 250°C (482°F) |
| SRV discharge line piping, from the vent wall penetration to the suppression pool surface | 2 | 3.72 MPa (540 psig) | 250°C (482°F) |

### 1.2.2.2 Controls and Instrumentation

#### 1.2.2.2.1 Rod Control and Information System

The Rod Control and Information System (RC&IS) is to safely and reliably provide:

- The capability to control reactor power level by controlling the movement of control rods in reactor core in manual, semiautomatic, and automated modes of plant operations.
- Controls for some RC&IS bypass and surveillance test functions, and summary information of control rod positions and status in the main control room.
- Transmission of fine motion control rod drive (FMCRD) status and control rod positions and status data to other plant systems (e.g., the Process Computer System).
- Automatic control rod run-in function of all operable control rods following a scram (scram follow function).
- Automatic enforcement of rod movement blocks to prevent potentially undesirable rod movements (these blocks do not have an effect on scram insertion function).
- Control capability for insertion of all control rods by an alternate and diverse method [alternate rod insertion (ARI) function].
- The capability to enforce a preestablished sequence for control rod movement when reactor power is below the low power setpoint.
- The capability to enforce fuel operating thermal limits when reactor power is above the low power setpoint.
- The capability to provide for Selected Control Rod Run In (SCRRI) function for mitigating a loss of feedwater heating event.

The RC&IS is classified as a non-safety-related system, it has a control design basis only, and is not required for the safe shutdown of the plant. A failure of the RC&IS will not result in gross fuel damage. However, the rod block function of RC&IS is important in limiting the consequences of a rod withdrawal error, and prevention of local fuel operating thermal limits violations during normal plant operations. Therefore, RC&IS is designed to be single-failure proof and highly reliable.

The RC&IS consists of several different types of cabinets (or panels), which contain special electronic/electrical equipment modules, and a dedicated operator interface on the main control panel in the MCR.

The RC&IS is redundant system that consists of two independent channels for normal control rod position monitoring and control rod movements. The two channels receive the same but separate input signals and perform the same exact functions. For normal functions of RC&IS, the two channels must always be in agreement and any disagreement between the two channels results in rod block. However, the protective function logic of RC&IS (i.e., rod block) is designed such that the detection of a rod block condition in only one channel of RC&IS would result in a rod block.

There are four types of electronic/electrical cabinets that make up the RC&IS. They are:

- Rod action control cabinets (RACC)
- Remote communication cabinets (RCCs)
- Fine motion driver cabinets (FMDCs)
- Rod brake controller cabinets (RBCCs)

In addition, RC&IS includes a fiber-optic dual channel multiplexing network that is used for transmission of rod position and status data from RCCs to the Rod Action and Position Information (RAPI), and rod block/movement command from RAPI to RCCs. A summary description of each of the above functions is provided below.

***Rod Action Control Cabinets (RACC)***

There are two RACCs in the control room; RACC Channel A and RACC Channel B that provide for a dual redundant architecture. Each RACC consists of three main functional subsystems, as follows:

- Automated Thermal Limit Monitor (ATLM)
- Rod Worth Minimizer (RWM)
- Rod Action and Position Information (RAPI)

***Remote Communication Cabinets (RCC)***

The remote communication cabinets (RCCs) contain a dual channel file control module (FCM) and several dual channel rod server modules (RSMs). The FCM interfaces with the RSMs and RAPI.

***Fine Motion Driver Cabinets (FMDC)***

The fine motion driver cabinets (FMDCs) consist of several stepping motor driver modules. Each stepping motor driver module contains an electronic converter/inverter that converts the incoming 3-phase ac power into dc and then inverts the dc power to variable voltage/frequency ac power that is supplied to FMCRD stepping motors. For each converter/inverter, there exists an inverter controller (IC) that controls the duration of power supplied to the stepping motors under the command of RSMs.

### *Rod Brake Controller Cabinets (RBCC)*

The rod brake controller cabinets (RBCCs) contain electrical power supplies, electronic (or relay) logic, and other associated electrical equipment for the proper operation of the FMCRD brakes. Signals for brake disengagement/engagement are received from the associated rod server modules. The brake controller logic provides two separate (channel A and channel B) brake status signals to the associated rod server module.

### *RC&IS Multiplexing Network*

The RC&IS multiplexing network consists of two independent channels (channel A and channel B) of fiber-optic communication links between the RACCs (channel A and channel B), and the dual channel file control modules located in the remote communication cabinets.

The plant essential multiplexing network interfaces with FMCRD redundant separation switches (A/B) and provides the appropriate status signals to the RACC that is used in the RC&IS logic for initiating rod block signals if a separation occurs. The essential multiplexing network is not part of the RC&IS scope.

### *RC&IS Power Sources*

RC&IS equipment derives its power from two different sources. FMDCs and RBCCs derive their power from the plant divisional power sources that are backed up by plant diesel generators. All other RC&IS equipment derive their power from the plant uninterruptible ac power system.

## 1.2.2.2.2 Control Rod Drive System

The Control Rod Drive (CRD) System is composed of three major elements:

- the fine motion control rod drive (FMCRD) mechanisms,
- the hydraulic control unit (HCU) assemblies, and
- the control rod drive hydraulic (CRDH) subsystem.

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. Simultaneous with scram, the FMCRDs also provide electric-motor-driven run-in of all control rods as a path to rod insertion that is diverse from the hydraulic-powered scram. The hydraulic power required for scram is provided by high pressure water stored in the individual HCUs. Each HCU is designed to scram up to two FMCRDs. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRDH Subsystem supplies high pressure demineralized water which is regulated and distributed to provide charging of the HCU scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the feedwater flow is not available.

During power operation, the CRD System controls changes in core reactivity by movement and positioning of the neutron absorbing control rods within the core in fine increments via the FMCRD electric motors, which are operated in response to control signals from the RC & IS.

The CRD System provides rapid control rod insertion (scram) in response to manual or automatic signals from the Reactor Protection System (RPS), so that no fuel damage results from any plant transient.

There are 177 FMCRDs mounted in housings welded into the RPV bottom head. Each FMCRD has a movable hollow piston tube that is coupled at its upper end, inside the reactor vessel, to the bottom of a control rod. The piston is designed such that it can be moved up or down, both in fine increments and continuously over its entire range, by a ball nut and ball screw driven at a nominal speed of 30 mm/sec by the electric stepper motor. In response to a scram signal, the piston rapidly inserts the control rod into the core hydraulically using stored energy in the HCU scram accumulator. The scram water is introduced into the drive through a scram inlet connection on the FMCRD housing, and is then discharged directly into the reactor vessel via clearances between FMCRD parts. The FMCRD scram time requirements with the reactor gauge pressure of 7.481 MPa (gauge) (1085 psig) as measured at the vessel bottom are:

| Percent Insertion | Time (sec) |
|---|---|
| 10 | ≤ 0.42 |
| 40 | ≤ 1.00 |
| 60 | ≤ 1.44 |
| 100 | ≤ 2.80 |

The FMCRD design includes an electro-mechanical brake on the motor drive shaft and a ball check valve at the point of connection with the scram inlet line. These features prevent control rod ejection in the event of a failure of the scram insert line. An internal housing support is provided to prevent ejection of the FMCRD and its attached control rod in the event of a housing failure. It uses the outer tube of the drive to provide support. The outer tube, which is welded to the drive middle flange, attaches by a bayonet lock to the base of the control rod guide tube section of the integral fuel support and control rod guide tube. The fuel support section, being supported by the lower core plate, in turn, prevents any downward movement of the drive.

The FMCRD is designed to detect separation of the control rod from the drive mechanism. Two redundant and separate Class 1E switches detect separation of either the control rod from the hollow piston or the hollow piston from the ball nut. Actuation of either switch will cause an immediate rod block and initiate an alarm in the MCR, thereby preventing the occurrence of a rod drop accident.

There are 89 HCUs, each of which provides sufficient volume of water stored at high pressure in a pre-charged accumulator to scram two FMCRDs at any reactor pressure. Each accumulator is connected to its associated FMCRDs by a hydraulic line that includes a normally-closed scram valve. The scram valve opens by spring action but is normally held closed by pressurized control air. To cause scram, the RPS provides a de-energizing reactor trip signal to the solenoid-operated pilot valve that vents the control air from the scram valve. The system is "fail safe" in that loss of either electrical power to the solenoid pilot valve or loss of control air pressure causes scram. The HCUs are housed in the safety envelope at the basemat elevation. This is a Seismic Category I structure, and the HCUs are protected from external natural phenomena such as earthquakes, tornados, hurricanes and floods, as well as from internal postulated accident phenomena. In this area, the HCUs are not subject to conditions such as missiles, pipe whip, or discharging fluids.

The CRDH subsystem design provides the pumps, valves, filters, instrumentation, and piping to supply the high pressure water for charging the HCUs and purging the FMCRDs. Two 100% capacity pumps (one on standby) supply the HCUs with water from the condensate treatment system and/or condensate storage tank for charging the accumulators and for supplying FMCRD purge water. The CRDH subsystem equipment is housed in the Seismic Category I portion of the reactor building to protect the system from floods, tornadoes, and other natural phenomena. The CRDH subsystem also has the capability to provide makeup water to the RPV while at high pressure as long as ac power is available.

The CRD System includes MCR indication and alarms to allow for monitoring and control during design basis operational conditions, including system flows, temperatures and pressures, as well as valve position indication and pump on/off status. Class 1E pressure instrumentation is provided on the HCU charging water header to monitor header performance. The pressure signals from this instrumentation are provided to the RPS, which will initiate a scram if the header pressure degrades to a low pressure setpoint. This feature assures the capability to scram and safely shut down the reactor before HCU accumulator pressure can degrade to the level where scram performance is adversely affected following the loss of charging header pressure.

Components of the system that are required for scram (FMCRDs, HCUs and scram piping), are classified Seismic Category I. The balance of the system equipment (pumps, valves, filters, piping, etc.) is classified as Seismic Category NS (non-seismic).

with the exception of the Class 1E charging water header pressure instrumentation, which is Seismic Category I. The major mechanical components are designed to meet ASME Code requirements as shown below:

| Component | ASME Code Class | Design Conditions | |
|---|---|---|---|
| | | Gauge Pressure | Temperature |
| FMCRD (RCPB parts) | 1 | 8.62 MPa (1250 psig) | 302°C (545°F) |
| Scram piping | 2 | 18.6 MPa (2700 psig) | 66°C (150°F) |
| HCU (scram related parts) | 2 | 18.6 MPa (2700 psig) | 66°C (150°F) |
| CRD pumps | Non-Code | 18.6 MPa (2700 psig) | 66°C (150°F) |
| CRDHS piping, valves | Non-Code | 18.6 MPa (2700 psig) | 66°C (150°F) |

The CRD System is separated both physically and electrically from the Standby Liquid Control System (SLCS).

#### 1.2.2.2.3 Feedwater Control System

The Feedwater Control System (FWCS) controls the flow of feedwater into the RPV to maintain the water level in the vessel within predetermined limits during all plant operating modes. The FWCS may operate in either single- or three-element control modes. At low reactor powers (when steam flow is either negligible or else measurement is below scale), the FWCS uses only water level measurement in single-element control mode. When steam flow is negligible, the Reactor Water Cleanup /Shutdown Cooling (RWCU/SDC) System overboard control valve can be controlled by the FWCS System in single-element mode in order to counter the effects of density changes during heatup and purge flows into the reactor. At higher powers, the FWCS in three-element control mode uses water level, main steamline flow, main feedwater line flow, and feedpump suction flow measurements for water level control.

The FWCS is a power generation (control) system with operation range between high water level and low water level trip setpoints. It is classified as non-safety-related. This system is not required for safety purposes, nor is it required to operate after the design basis accident. This system is only required to operate in the normal plant environment, and for power generation purposes only.

Reactor vessel narrow range water level is measured by three identical, independent sensing systems. For each level measurement channel, a differential pressure transmitter senses the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the reactor vessel. The FWCS uses microprocessor-based fault tolerant digital controllers (FTDCs) which will determine one validated narrow range level signal using the three level measurements as inputs to a signal validation algorithm. The validated narrow range water level is indicated on the main control console in the MCR.

Steam flow is sensed at the RPV MSL nozzle venturi's in each of the two main steamlines. The Multiplexing System signal conditioning algorithms process the venturi differential pressures and provide steam flow rate signals to the FTDCs for validation. These validated measurements are summed in the FTDCs to give the total steam flow rate out of the vessel. The total steam flow rate is indicated on the main control console in the MCR.

Feedwater flow is sensed at a single flow element in each of the two feedwater lines. The Multiplexing System signal conditioning algorithms process the flow element differential pressure and provide feedwater flow rate signals to the FTDCs. These validated measurements are summed in the FTDCs to give the total feedwater flow rate into the vessel. The total feedwater flow rate is indicated on the main control console in the MCR.

Feedpump suction flow is sensed at a single flow element upstream of each feedpump. The Multiplexing System signal conditioning algorithms process the flow element differential pressure and provide the suction flow rate measurements to the FTDCs. The feedpump suction flow rate is compared to the demand flow for that pump, and the resulting error is used to adjust the actuator in the direction necessary to reduce that error. Feedpump speed change and low flow control valve position control are the flow adjustment techniques involved.

Three modes of feedwater flow control (and, thus, level control) are provided: (1) single-element control; (2) three-element control; and (3) manual control. Each FTDC will execute the control software for all three of the control modes. Actuator demands from the redundant FTDCs will be sent over the Multiplexing System to field voters which will determine a single demand to be sent to each actuator. Each feedpump speed or control valve position demand may be controlled either automatically by the control algorithms in the FTDCs or manually from the main control room through the FTDCs.

Three-element automatic control is provided for normal operation. Three-element control uses water level, feedwater flow, steam flow, and feedpump flow signals to determine the feedpump demands. The total feedwater flow is subtracted from the total steam flow signal, yielding the vessel flow mismatch. The flow mismatch, summed with the conditioned level error from the master level controller, provides the demand for the master flow controller. The master flow controller output provides the demand signal to the adjustable speed drives (ASD) for each feedpump.

In the single-element control mode, only conditioned level error is used to determine the feedpump demand. The master level controller conditions the level error and sends it directly to the feedpump ASDs, and/or low flow control valve actuator. When the reactor water inventory must be decreased (e.g., during very low steam flow rate conditions), the RWCU/SDC System overboard control valve is controlled by the FWCS in single-element control. Reactor water is discharged through the RWCU/SDC System to the condenser.

Each feedpump flow control actuator can be controlled 'manually' from the main control panel by selecting the manual mode for that feedpump. In manual mode, the operator may increase or decrease the demand that is sent directly to the ASD of the chosen feedpump.

The FWCS also provides interlocks and control functions to other systems. When the reactor water level reaches the high level trip setpoint, the FWCS simultaneously annunciates an alarm in the MCR, sends a trip signal to the turbine control system to trip the turbine generator, sends trip signals to all feedpumps, and closes the main feedwater discharge valves. This interlock is enacted to protect the turbine from damage from high moisture content in the steam caused by excessive carryover, while preventing water level from rising any higher.

The FWCS sends a signal to the main steamline condensate drain valves to open when steam flow rate is below a pre-determined setpoint. This also protects the turbine from damage caused by excessive moisture in the steamline.

Feedwater flow is delivered to the reactor vessel through a combination of three adjustable speed motor-driven feedpumps and a low flow control valve. The low flow control valve (LFCV) is provided in the high-pressure feedwater heater bypass line. The LFCV can also be controlled by the manual/automatic transfer station which is part of the Condensate and Feedwater System.

The FWCS is powered by redundant uninterruptible power supplies (UPS). No single power failure will result in the loss of any FWCS functions.

Controllers to be used for the FWCS are triplicated, fault tolerant digital type with self-test and diagnostic capabilities.

### 1.2.2.2.4 Standby Liquid Control System

The Standby Liquid Control System (SLCS) provides an alternate method of reactor shutdown from full power to cold subcritical by the injection of a neutron absorbing solution into the RPV.

The SLCS interfaces with Class 1E 125 Vdc divisional power for the squib-type injection valves; for the valve which isolates the accumulator after injection; for accumulator solution level measurement, trip, and alarm functions; and for the particular NBS instrumentation and SSLC control logic which generates the anticipated transient without scram (ATWS) signal for automatic SLCS initiation.

The SLCS includes piping, valves, accumulator, and instrumentation designed to inject a neutron absorber solution into the reactor. The system is designed to operate over the range of reactor pressure conditions up to the elevated pressures of an ATWS event, and to inject sufficient neutron absorber solution to reach hot subcritical conditions after system initiation.

Instrumentation is provided to the operator for monitoring the status of the SLCS, and for alarming any off standard condition.

### 1.2.2.2.5 Neutron Monitoring System

The Neutron Monitoring System (NMS) provides indication of neutron flux in the core in all modes of reactor operation. The safety-related NMS functions are the startup range neutron monitor (SRNM), the local power range monitor (LPRM), and the average power range monitor (APRM). The non-safety-related subsystem is the automated fixed in-core probe (AFIP). The LPRMs and APRMs make up the power range neutron monitor (PRNM) subsystem. The safety-related portions of the NMS are classified, Seismic Category I, and IEEE Class 1E.

The NMS provides signals to the RPS, the RC&IS, and the Process Computer System. The NMS provides trip signals to the RPS for reactor scram on rising excessive neutron flux or too short a period for flux generation.

The NMS consists of four divisions which correspond and interface with those of the RPS, and this independence and redundancy assure that no single failure will interfere with the system operation.

The SRNM subsystem is comprised of eight SRNM channels which are divided into four divisions and independently assigned to three bypass groups such that up to three SRNM channels are allowed to be bypassed at any time while still providing the required monitoring and protection capability.

The LPRM function of the PRNM subsystem is comprised of 21 LPRM assemblies evenly distributed throughout the cross-section of the core. There are four LPRM detectors within each LPRM assembly, evenly spaced from near the bottom of the fuel region to near the top of the fuel region. These 84 detectors are assigned to four sets of 21 detectors each. The signals from each set of 21 LPRM detectors are assigned to one APRM channel, with these signals summed and averaged to form a partial APRM signal. This partial APRM signal is transmitted to the other three APRM channels through electrical isolation. Within each of the four APRM channels, all four partial APRM signals are then averaged to form a final APRM signal. The partial APRM signal transmission between divisions is carried out through fiber optic pathways which serve as effective electrical isolation devices. Electrical and physical separation of the division is thus maintained and optimized to satisfy the safety-related system requirement. With the four divisions, redundancy criteria are met since a scram signal can still be initiated with a postulated single failure under allowed APRM bypass conditions.

All the NMS instruments are primarily based on the digital measurement and control (DMC) design practices that use digital design concepts. All NMS DMC instruments follow a modular design concept such that each modular unit or its subunit is replaceable upon repair service.

The SRNM subsystem covers the lower power range from the source range ($1 \times 10^3$ nv) to 15% of rated reactor power. The PRNM subsystem overlaps the SRNM, covering the range from approximately 1% to 125% of rated reactor power.

The AFIP subsystem is comprised of sensors and their associated cables, as well as the signal processing electronic unit. The AFIP sensors are the gamma thermometer type. There are four AFIP gamma thermometer sensors evenly distributed across each LPRM assembly, with one gamma thermometer installed next to each LPRM detector. Consequently, there are AFIP sensors at all LPRM locations. The AFIP sensor cables are routed within the LPRM assembly and then out of the RPV through the LPRM assembly penetration to the vessel. The AFIP System generates signals proportional to the axial power distribution at the radial core locations of the LPRM detector assemblies. The AFIP signal range is sufficiently wide to accommodate the corresponding local power range that covers from 0% to 125% of reactor rated power.

The AFIP gamma thermometer sensor has a constant or very stable detector sensitivity that will not significantly change due to radiation exposure or other reactor conditions. The AFIP gamma thermometer, however, can be calibrated by using a built-in calibration device inside the gamma thermometer/LPRM assembly. Due to its stable sensitivity and rugged hardware design, the AFIP sensor has a lifetime much longer than that of the LPRM detectors. The AFIP sensors in an LPRM assembly are replaced together with the LPRM detectors when the whole LPRM assembly is replaced.

#### 1.2.2.2.6 Remote Shutdown System

The Remote Shutdown System (RSS) provides the means to safely shut down the reactor from outside the main control room. The RSS provides remote manual control of the systems necessary to: (a) achieve prompt hot shutdown of the reactor after a scram, (b) achieve subsequent cold shutdown of the reactor, and (c) maintain safe conditions during shutdown.

The remote shutdown system is classified as a non-safety-related system. The RSS does not include control interfaces with nuclear safety-related equipment.

To achieve a safe and orderly plant shutdown from outside the main control room, controls and indicators necessary for operation of the following system and equipment are provided on the remote shutdown panel.

- Reactor Water Clean-up/Shutdown Cooling (RWCU/SDC) System
- Control Rod Drive (CRD) System (makeup function)
- Reactor Component Cooling Water (RCCW) System
- Plant Service Water (PSW) System
- Electrical Power Distribution System
- Nuclear Boiler System (NBS) instrumentation
- Reactor Building HVAC

#### 1.2.2.2.7 Reactor Protection System

The Reactor Protection System (RPS) initiates an automatic and prompt reactor trip (scram) by means of rapid hydraulic insertion of all control rods whenever selected plant variables exceed preset limits. The primary function is to effect a reactor shutdown before fuel damage occurs. The RPS also provides reactor status information to other systems and will cause an alarm annunciation in the MCR whenever selected plant variables approach the preset limits.

The RPS is a safety protection system, differing from a reactor control system or a power generation system. The RPS and its components are safety-related. The RPS and the system electrical equipment are classified as Seismic Category I and IEEE Class 1E.

Basic system parameters are:

| | |
|---|---|
| Number of independent divisions of equipment | 4 |
| Minimum number of sensors per trip variable (at least one per division) | 4 |
| Number of automatic trip systems (one per division) | 4 |
| Automatic trip logic used for plant sensor inputs (per division) | 2-out-of-4 |
| Separate automatic trip logic used for division trip outputs | 2-out-of-4 |
| Number of separate manual trip systems | 2 |
| Manual trip logic | 2-out-of-2 |

The RPS initiates reactor trip signals within individual sensor channels when any one or more of the conditions listed below exists during reactor operation. Reactor scram will result if system logic is satisfied.

- Drywell pressure high
- Reactor power (neutron flux or simulated thermal power) exceeds limits for operating mode
- Reactor power rapid increase
- Reactor vessel pressure high
- Reactor water level low (Level 3)
- Reactor water level high (Level 8)
- Main steam isolation valves closed (Run mode only)
- Control rod drive charging header pressure low
- Suppression pool temperature high
- Operator-initiated manual scram

The RPS is an overall complex of instrument channels, trip logic, trip actuators, manual controls, and scram logic circuitry that initiates the rapid insertion of control rods by hydraulic force to scram the reactor when unsafe conditions are detected. The RPS uses the functions of the essential multiplexing subsystem (EMS) and the SSLC system to perform its functions.

The RPS is divided into four redundant divisions of sensor channels, trip logics, and trip actuators, and two divisions of manual scram controls and logic circuitry. Each division has a separate IEEE Class 1E power supply taken from the safety-related UPS 120 Vac power supply. The automatic and manual scram initiation logic systems are independent of each other and use diverse methods and equipment to initiate a reactor scram. The RPS design is such that, once a full reactor scram has been initiated automatically or manually, this scram condition seals-in such that the intended fast insertion of control rods into the reactor core can continue to completion. After a time delay, the design requires operator action to reset the scram logic to the untripped state.

The RPS scram logic circuits are arranged so that coincident trips in two of the four divisions (2-out-of-4 logic) of sensor channels and in two of the four trip system outputs to the actuating devices are required to effect a scram. This arrangement permits a single failure in one division to occur without either causing a scram or preventing the other three divisions from causing a scram. For example, the single failure may be in either system logic or the individual power supply for that division.

Each logic division and its associated power supply is separated both physically and electrically from the other divisions. This arrangement permits one division at a time to be taken out of service (bypassed) for testing during reactor operation. The other divisions then perform the RPS function with system logic in a 2-out-of-3 arrangement.

#### 1.2.2.2.8 Automatic Power Regulator System

The Automatic Power Regulator (APR) System is classified as a power generation system and is not required for safety. Events requiring control rod scram are sensed and controlled by the safety-related RPS, which is completely independent of the APR System.

The APR System controls reactor power during reactor startup, power generation, and reactor shutdown by appropriate commands to change rod positions. The APR system also controls the pressure setpoint or turbine bypass valve position during reactor heatup and depressurization (e.g., to control the reactor cool down rate). The APR System consists of redundant process controllers. Automatic power regulation is achieved by appropriate control algorithms for different phases of reactor operation which include approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and reactor depressurization and cooldown. The APR System receives input from the Neutron Monitoring System, the Process Computer System, the power generation control subsystem, the Steam Bypass and Pressure Control System, and the operator's control console. The output demand signals from the APR System are sent to the RC & IS to position the control rods, and to the Steam Bypass and Pressure Control System for automatic load following operations. The power generation control subsystem performs the overall plant startup, power operation, and shutdown functions. The APR System performs those functions associated with reactor power changes and with reactor pressure controller setpoint (or turbine bypass valve position) changes during reactor heatup or depressurization.

The automatic power regulation system control functional logic is performed by redundant, microprocessor-based fault-tolerant digital controllers (FTDC). The FTDC performs many functions. It reads and validates inputs from the non-essential multiplexing system (NEMS). It performs the specific power control calculations and processes the pertinent alarm and interlock functions, then updates all system outputs to the NEMS. To prevent computational divergence among the redundant processing channels, each channel performs a comparison check of its calculated results with other redundant channels. The internal FTDC architecture features redundant multiplexing interfacing units for communications between the NEMS and the FTDC processing channels.

During normal operation, the APR System interfaces with the operator's console to perform its desired functions. The operator's control panel for automatic plant startup, power operation, and shutdown functions is part of the power generation control subsystem. The power generation control subsystem initiates demand signals to various controllers to carry out the pre-defined control functions. The functions associated with reactor power control are performed by the APR System. For reactor power control, the APR System contains algorithms that can change reactor power by control rod motions. During automatic load following operation, the APR System interfaces with the Steam Bypass and Pressure Control System to coordinate main turbine and reactor power changes to accomplish load following.

The normal mode of operation for the APR System is automatic. If any system or component conditions are abnormal during execution of the prescribed sequences of operation, the power generation control subsystem will be automatically switched into the manual mode and the operator can manipulate control rods using the normal controls. A failure of the APR System will not prevent manual control of the reactor, nor will it prevent safe shutdown of the reactor.

The APR System digital controllers are powered by redundant uninterruptible non-Class 1E power supplies and sources. No single power failure will result in the loss of any APR System function.

#### 1.2.2.2.9 Steam Bypass & Pressure Control System

The Steam Bypass & Pressure Control (SB&PC) System is a non-safety-related system whose design objective is to enable a fast and stable response to pressure and system disturbances, and to setpoint changes, over the operating range using turbine control valves and turbine bypass valves for controlling pressure. In addition, the design objective of the SB&PC System is to discharge reactor steam directly to the main condenser to regulate reactor pressure whenever the turbine cannot use all of the steam generated by the reactor.

The SB&PC System is designed to control reactor pressure during plant startup, power generation and shutdown modes of operation. This is accomplished through control of the turbine control valves and/or turbine bypass valves, such that susceptibility to reactor trip, turbine-generator trip, MSIV closure and SRV opening is minimized.

Command signals for the turbine control valves and the turbine bypass valves are generated by a triplicated FTDC using feedback signals from RPV pressure signals. For normal operation, the turbine control valves regulate steam pressure. However, whenever the total steam flow demand from the pressure controller exceeds the effective turbine control valve steam flow capability, the SB&PC System sends the excess steam flow directly to the main condenser, through the turbine bypass valves.

The SB&PC System functional logic and process control functions are performed by triplicated microprocessor-based FTDC similar to controllers used in FWCS. Because of the triple redundancy, it is possible to lose one complete processing channel without impacting the system function. This also facilitates taking one channel out of service for maintenance or repair while the system is on-line. The SB&PC system receives input signals from other systems and sensors as follows:

- turbine bypass valve position switches;
- turbine bypass valve servo current sensors;
- Turbine Control System (TCS) turbine trip sensors;
- TCS power/load unbalance relay operation;
- Turbine Bypass System (TBS) hydraulic power supply trouble sensors;
- NBS MSIV position switches;
- NBS narrow and wide range dome pressure transmitters;
- main condenser low vacuum sensors; and
- operator manual commands and manual switch positions.

The SB&PC system provides output signals to:

- turbine bypass valves
- turbine control valves

- APR System;
- various related control room indicators and alarms; and
- process computer.

At steady-state plant operation, the SB&PC System maintains reactor vessel pressure at a nearly constant value, to ensure optimum plant performance. During normal operational plant maneuvers (pressure setpoint changes, level setpoint changes), the SB&PC System provides responsive, stable performance to minimize vessel water level and neutron flux transients. During plant startup and heatup, the SB&PC System provides for automatic control of the reactor pressure. Independent control of reactor pressure and power is permitted during reactor-vessel heatup, by varying turbine bypass flow as the main turbine is brought up to speed and synchronized.

Additional reactor system pressure control functions are provided by other systems when the MSIVs are closed.

#### 1.2.2.2.10 Process Computer System

The Process Computer System (PCS) is a non-safety-related system. Its purpose is to promote efficient plant operation by:

- Performing the functions and calculations necessary for the evaluation of plant operation
- Providing a permanent historical record for plant operating activities and abnormal events
- Providing analysis, evaluation and recommendation capabilities for start-up, normal operation, safe plant shutdown and abnormal operating and emergency conditions
- Providing control and display capability on the main control room video display units.
- Providing the ability to directly control certain non-safety-related plant equipment through on-screen technology.

All division to division and safety-related to non-safety-related interfacing circuits are made up of fiber optic cables, which act as optical isolators for electrical separation. All power to the PCS is supplied by a non-safety-related redundant, uninterruptible power supply. No single power failure will cause the loss of any PCS function.

The PCS has self-checking provisions. It performs diagnostic checks to determine the operability of certain portions of the system hardware and performs internal programming checks to verify that input signals and selected program computations are either within specific limits or within reasonable bounds.

The PCS is composed of two subsystems; the performance monitoring and control subsystem (PMCS) and the power generation control subsystem (PGCS).

### Performance Monitoring and Control System

The PMCS is a set of software routines for the PCS input/output modules and various CPUs to supply various functions and calculations. The basic input types are as follows:

- Various analog pressure signals from sensors on or in the RPV, the drywell, individual equipment and the various plant buildings.
- Various analog temperature signals from sensors on or in the RPV, the drywell, individual equipment and the various plant buildings.
- Various analog coolant and steam flow signals from sensors on or in the various pumps and pipes throughout the plant.
- Various digital "on/off" and "open/closed" signals from various switches and valve controllers throughout the plant.
- Various operator requests as input through the various consoles.

The basic output types are as follows:

- plant operating conditions;
- process trends;
- alarms;
- results of performance calculations;
- operator requests; and
- switchyard operating conditions

The types of calculations performed include but are not limited to the following:

- reactor core performance calculation;
- plant performance calculation;
- plant efficiency;
- turbine generator efficiency;

- condenser performance (thermal load and cleanliness);
- feedwater heater performance;
- moisture separator performance; and
- condensate demineralizer performance.

The function types performed in addition to the calculations include but are not limited to the following:

- data accumulation;
- indication of control rod position; and
- surveillance test guide.

***Power Generation Control Subsystem***

The PGCS is a a set of software routines residing on the Process Computer System which produce control outputs for the automated control sequences associated with plant start-up, shutdown, and normal power generation. The PGCS receives the same type inputs as described for the PMCS control commands and sends system mode change and set-point change commands to subloop controllers to support the plant automation features. The automation process is divided into phases corresponding to plant start-up, shutdown, and normal power generation. Each phase is then divided into several break-points, or logical steps in plant operation. Automation proceeds under PGCS control until the end of a break-point division is reached, at which time the operator must confirm that conditions are acceptable before automation sequence can continue.

### 1.2.2.2.11 Refueling Machine Computer

The Refueling Machine is designed for automatic operation by a programmed computer operated from a console above the refueling floor.

The computer will control all direct refueling machine movements to any selected core location through the established XYZ coordinate system.

### 1.2.2.2.12 Leak Detection and Isolation System

The Leak Detection and Isolation System (LD&IS) detects and monitors leakage from the containment, preventing the release of radiological leakage from the reactor coolant boundary. The system initiates safety isolation functions by closure of inboard and outboard containment isolation valves.

The following functions are provided by LD&IS:

- containment isolation following a LOCA event;
- main steam lines isolation;
- isolation condenser system process lines isolation;
- RWCU/SDC system process lines isolation;
- fuel and auxiliary pools cooling system process lines isolation;
- reactor component cooling water lines to DW coolers isolation;
- drywell sumps liquid drain lines isolation;
- containment purge and vent lines isolation;
- reactor building HVAC air exhaust ducts isolation;
- CRD charging and purge water lines isolation;
- fission products sampling line isolation;
- monitoring of identified and unidentified leakages in the drywell;
- monitoring of condensate flow from the drywell air coolers;
- monitoring of the vessel head flange seal leakage;
- monitoring of valve stems leakages in the containment.

The following leakage detection functions are provided by other plant systems:

- Monitoring of fission products in the drywell;
- Monitoring of FMCRD leakage;
- Monitoring of plant sump levels and flow rates;
- SRV Steam Discharge.

The LD&IS monitors plant parameters such as flow, temperature, pressure, water level, etc., which are used to alarm and initiate the isolation functions.

At least two parameters are monitored for an isolation function. The signal parameters are processed by the Safety System and Logic Control system (SSLC) which generates the trip signals for initiation of isolation functions.

The LD&IS safety-related functions have four divisional channels of sensors for each parameter. Two-out-of-four coincidence voting within a channel is required for initiation of the isolation function. The control and decision logic are of fail-safe design which assures isolation on loss of power. The logic is energized at all times and de-energizes to trip for isolation function.

Loss of one divisional power or one monitoring channel will not cause inadvertent isolation of the containment. Different divisional isolation signals are provided to the inboard and outboard isolation valves.

The LD&IS is designed to allow periodic testing of each channel to verify it is capable to perform the intended function.

LD&IS is a safety-related system and is classified Seismic Category I.

The LD&IS initiates isolation functions automatically. All isolation valves have individual manual control switches and valve position indication in the MCR. However, the isolation signal overrides any manual control to close the isolation valves.

Manual control switches in the control logic provide a backup to automatic initiation of isolation as well as capability for reset, bypass and test of functions.

The monitored plant parameters are measured and recorded by the Process Computer System, and are displayed on demand. The abnormal indications and initiated isolation functions are alarmed in the MCR.

#### 1.2.2.2.13 Safety System Logic and Control System

The Safety System Logic and Control (SSLC) System provides the decision logic facility for implementing safety-related logic functions. These functions enable the safety-related systems to perform their plant protection tasks.

The SSLC performs the following functions:

- Sensor channel trip decisions
- System coincidence trip decisions (2-out-of-4 logic or 2-out-of-3 logic)
- Control and interlock logic
- ATWS prevention and mitigation
- Manual division trip and isolation
- Division-of-sensors bypass

- Division maintenance bypass (division out-of-service)
- Calibration and self-diagnosis

The SSLC System is configured as a four-division data acquisition and control system, with each division containing an independent set of microprocessor-based, software-controlled logic processors. The four divisions exchange data via fiber optic data links to implement cross-channel data comparison.

The SSLC System acquires data from redundant sets of sensors of the interfacing safety-related systems and provides control outputs to the final component actuators. Data is received from the essential multiplexing system (EMS) or directly hardwired from transmitters or sensors.

### 1.2.2.3 Radiation Monitoring Systems

#### 1.2.2.3.1 Process Radiation Monitoring System

The primary functions of the Process Radiation Monitoring System (PRMS) are to:

- monitor and record the various gaseous and liquid process streams and effluent releases;
- initiate alarms in the main control room to warn operating personnel of high radiation activity; and
- initiate the appropriate safety actions and controls to prevent further radioactivity releases to the environment.

This system provides both safety-related and non-safety-related instrumentation for radiological monitoring, sampling and analysis of identified process and effluents streams throughout the plant.

The process and effluent paths and/or areas as described herein are monitored for potential high radioactivity releases. The radiation monitors of the first six items below are safety-related Class 1E instrumentation, while the remaining of the PRMS monitors are considered non-safety-related provided to monitor plant operations.

- Main steam line (MSL) tunnel area — 2 divisional channels

  The MSL tunnel area is continuously monitored for high gross gamma radioactivity in the steam flow to the turbine. Shutdown of the main condenser vacuum pump is automatically initiated on any 1-out-of-2 channel trip.

- Reactor building safety envelope ventilation exhaust — 4 divisional channels

The air vent exhaust from the safety envelope is continuously monitored for gross gamma radioactivity. On high level the containment ventilation ducts are isolated on any 2-out-of-4 channel trip.

- Containment purge exhaust—4 Divisional Channels

  The radiation level in the purge exhaust from the containment is monitored for gross gamma radioactivity. On a high level, the ventilation ducts to the containment are automatically isolated on any 2-out-of-4 channel trip.

- Refueling area ventilation exhaust — 4 divisional channels

  The air vent exhaust from the refueling area is continuously monitored for gross gamma radioactivity. On high level, the ventilation ducts in this area are isolated on any 2-out-of-4 channel trip.

- Control room envelope air intake supply — 4 divisional channels

  The air intake to the MCR envelope area is continuously monitored for gross gamma radioactivity. On high level, the MCR ventilation ducts are isolated and the emergency air circulation system is activated on any 2-out-of-4 channel trip.

- Isolation condenser vent exhaust—4 Divisional Channels

  The atmospheric pool area within the confines of each isolation condenser is continuously monitored for gross gamma radioactivity. On high radiation level, the affected isolation condenser is automatically isolated through closure of the steam line and condensate return line isolation valves.

- Turbine building ventilation exhaust — 1 channel

  The air vent exhaust from the Turbine Building is continuously sampled through an isokinetic probe and monitored for airborne radioactivity by a beta/gamma sensitive detector and filters for collecting air particulates and iodine. A tritium monitor is also provided for sample collection. Alarms are initiated on high radiation and on abnormal sampling flow.

- Charcoal vault ventilation exhaust — 1 channel

  The vent exhaust from the charcoal vault is continuously monitored for gross gamma radioactivity that may result from leaks in the charcoal beds. An alarm is initiated on high radiation.

- Pre-treated main condenser offgases — 1 channel

  The pre-treated main condenser offgases are continuously sampled and monitored for gross gamma radioactivity. Alarms are initiated on high radiation and on abnormal sampling flow. Vial sampling is provided for periodic isotopic analysis.

- Post treated main condenser offgases — 2 channels

  The treated off-gases are continuously sampled and monitored for airborne radioactivity by two gas samplers and filters for collecting air particulates and halogens. Each gas sampler consists of a beta/gamma sensitive detector and a source check for periodic testing. On high radiation, the offgases are routed through the entire charcoal bed for holdup. On extremely high radiation, the offgas discharge to the stack is automatically isolated. Alarms are initiated on high radiation levels and on abnormal sampling flow. Vial sampling is provided for periodic isotopic analysis.

- Plant stack discharge — 2 channels

  The discharge through the stack is continuously sampled through an isokinetic probe and monitored for airborne radioactivity by two separate channels. Each channel consists of a beta/gamma sensitive detector with a source check and a high-range ion chamber. In addition, filters are provided for collecting air particulate and halogens, and components are provided for collecting and sampling tritium. Alarms are initiated on high radiation levels and on abnormal sampling flow.

- Radwaste building ventilation exhaust—1 channel

  The air vent exhaust from the radwaste building is continuously sampled through an isokinetic probe and monitored for airborne radioactivity by a beta/gamma sensitive detector with a source check and filters for collecting air particulates and iodine. A tritium monitor is also provided for sample collection. Alarms are initiated on high radiation and on abnormal sampling flow.

- Radwaste liquid discharge — 1 channel

  The liquid waste discharge from the plant is continuously sampled and monitored by a liquid sampler consisting of a scintillation detector and a source check. Alarms are initiated on high radiation levels and on abnormal sampling flow. On extremely high radiation in the discharged waste, the flow is automatically terminated and isolated.

- Drywell sump liquid discharge — 2 channels, 1 channel per sump

The liquid discharge from each of the two drywell sumps is monitored by an in-line ion chamber. On high radiation, the discharge to the radwaste building is terminated and alarmed.

- Turbine gland steam condenser discharge — 1 channel

  The discharge from the main turbine gland steam condenser is continuously monitored for airborne radioactivity by a digital gamma ventilation detector. An alarms is initiated on high radiation level.

- Intersystem radiation leakage — 2 channels, 1 channel per RCCW system loop

  Intersystem leakage into each loop of the Reactor Closed Cooling Water System is monitored by an in-line scintillation detector for gross gamma radioactivity. An alarm is initiated on high radiation.

- Reactor building ventilation exhaust —1 channel

  The air vent exhaust from the Reactor Building is continuously sampled through an isokinetic probe and monitored for airborne radioactivity by a beta/gamma sensitive detector with a source check and filters for collecting air particulates and iodine. A tritium monitor is also provided for sample collection. Alarms are initiated on high radiation and on abnormal sampling flow.

- Fission Products Releases —3 channels

  The atmosphere in the drywell is sampled and monitored for gross gamma radioactivity resulting from fission products releases. One channel monitors for noble gases, another channel monitors for air particulates, and the third channel monitors for halogens. Alarms are activated in the main control room on high radiation levels.

#### 1.2.2.3.2 Area Radiation Monitoring System

The Area Radiation Monitoring (ARM) System continuously monitors the gamma radiation levels within the various areas of the plant and provides an early warning to operating personnel when high radiation levels are detected so the appropriate action can be taken to minimize occupational exposure.

The ARM System is composed of multiple channels which utilize gamma sensitive detectors, associated digital radiation monitors, auxiliary units, and local audible warning devices. Each monitor has two adjustable trip circuits for alarm initiation, one high radiation level trip and one downscale trip. Also, each radiation monitor will actuate an alarm on loss of power or when gross equipment failure occurs.

The gross gamma radiation levels are monitored on a continuous basis, to signal any change in exposure rates which may be caused by operational transients, maintenance activities, or inadvertent release of radioactivity. Plant operating personnel are warned of any high radiation level by MCR alarms as well as audible area alarms.

The system monitoring range covers a span from $10^{-6}$Gy to $10^{2}$ Gy ($10^{-2}$ mR/hr to $10^{4}$ R/hr).

This system is non-safety-related. The radiation monitors are powered from the non-Class 1E vital 120 Vac source which is available continuously and during loss of site power.

The trip alarm setpoints will be established in the field following equipment installation at the site. The exact settings will be based on sensor location, background radiation levels, expected radiation levels, and low occupational radiation exposures.

### 1.2.2.3.3 Containment Atmospheric Monitoring System

The primary function of the Containment Atmospheric Monitoring System (CAMS) is to monitor the atmosphere in the containment for high gross gamma radiation levels and for high concentration levels of oxygen and hydrogen during post accident conditions. These three parameters are also monitored during normal reactor operations. The atmosphere in the drywell and in the suppression chamber is monitored and sampled by two independent, redundant CAMs subsystems.

CAMS is manually activated during normal plant operation to start the radiation monitoring and gas sampling process. For post accident monitoring, CAMS is automatically activated to perform its monitoring functions. The area of sampling can be manually selected or sequentially controlled between the drwell and the wetwell.

CAMS is a two-division monitoring system comprising two radiation monitoring channels per division and a gas sampling and analyzer rack per division. Radiation monitoring and gas sampling are provided for the drywell and for the airspace above the suppression pool. One gamma sensitive ion chamber and one digital log radiation monitor are used by each radiation monitoring channel. Two channels each for CAMS A & B are provided to monitor radiation levels in the containment. The radiation monitoring range is $10^{-2}$ Gy/hr to $10^{5}$ Gy/hr (1 R/hr to $10^{7}$ R/hr).

In the post accident operational mode, the safety function of CAMS is to continuously sample the oxygen and hydrogen contents in the containment, and display the results in the main control room. This information is then used by the operator to assess containment integrity and initiate flammability control if CAMS indicates the presence of a potentially explosive gas mixture in the containment.

Alarms and digital readouts are provided in the MCR for indications of high radiation dosage rates, inoperative radiation monitors, high oxygen levels, high hydrogen levels and of abnormal sampling for each subsystem. Each gas sampling rack is provided with its own gas calibration sources of known concentration levels to calibrate periodically the oxygen and hydrogen analyzers and sensors. Each oxygen and hydrogen gas sampling channel is checked for proper calibration and response at two or more input gas levels, one at the zero gas concentration level and the other at a nominal level from the calibrated gas sources.

CAMS is classified as a safety-related system and Seismic Category I. Power to each subsystem is provided from uninterruptible Class 1E 120 Vac divisional sources.

### 1.2.2.4 Core Cooling Systems

#### 1.2.2.4.1 Reactor Water Cleanup/Shutdown Cooling System

See discussion in ~~Subsection 1.2.2.4.1~~Subsection 5.4.8.

#### 1.2.2.4.2 Isolation Condenser System

The Isolation Condenser System (ICS) removes decay heat after any reactor isolation during power operations. Decay heat removal limits further pressure rises and keeps the RPV pressure below the SRV pressure setpoint. It consists of three independent loops, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating/evaporating water in the IC/PCC pool which is vented to the atmosphere.

The ICS is initiated automatically on either a high reactor pressure, or MSIV closure, or a Level 2 signal. To start an IC into operation, the motor-operated condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually by the operator from the MCR. A pneumatic-operated condensate return bypass valve is provided for each IC which opens if the 125 Vdc power is lost.

The ICS is isolated automatically when either a high radiation level or excess flow is detected in the steam supply line or condensate return line.

The IC/PCC pool is divided into subpools which are interconnected at their lower ends to provide full use of the water inventory for heat removal by any IC. The IC/PCC pool is normally cooled by the FAPCS. During IC operation IC/PCC pool water will boil, and the steam produced will be vented to the atmosphere. This boil-off action of nonradioactive water is a safe means for removing and rejecting all reactor decay heat.

The IC/PCC pool has an installed capaci that provides at least 72 hours of reactor decay heat. The heat rejection process car e continued indefinitely by replenishing the IC/PCC pool inventory. If normal make-up systems are unavailable, make-up can be provided via post-LOCA pool water make-up connections located just above grade level outside the reactor building. These lines are classified Quality Group C and Seismic Category I. This make-up can be accomplished without any valving changes in the reactor building no matter what the prior operating mode of the FAPCS might have been.

The ICS passively removes sensible and core decay heat from the reactor (i.e., heat transfer from the IC tubes to the surrounding IC/PCC pool water is accomplished by natural convection, and no forced circulation equipment is required) when the normal heat removal system is unavailable following any of the following events:

- reactor isolation at power operating conditions;
- reactor hot standby mode; and
- during station blackout (i.e., unavailability of all ac power).

The ICs are sized to remove post-reactor isolation decay heat with two out of three ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions with occasional venting of radiolytically generated noncondensable gases to the suppression pool. Since the heat exchangers (ICs) are independent of station ac power, they will function whenever normal heat removal systems are unavailable, to maintain reactor pressure and temperature below limits.

The heat removal capacity of the ICS (with two of three IC loops in service) is at least 60 MWt (each ICS is designed for 30 MWt capacity and is comprised of two identical modules), at a reactor gauge pressure of 7.240 MPa (1050 psig) with saturated steam.

The portions of the ICS (including isolation valves) which are located inside the containment and on the steam lines out to the IC flow restrictors are designed to ASME Code Section III, Class I. Other portions of the ICS are ASME Code Section III, Class 2. The IC pool is safety-related and Seismic Category I.

Periodic surveillance testing of the ICS valves can be performed by the control room operator via remote manual switches that actuate the isolation valves and the condensate return valves. The opening and closure of the valves is verified by their status lights.

### 1.2.2.4.3 Emergency Core Cooling System — Gravity-Driven Cooling System

Emergency core cooling is provided by the Gravity-Driven Cooling System (GDCS) in conjunction with the ADS in case of a LOCA. When a Level 1 signal is received, the ADS will depressurize the reactor vessel and the GDCS will inject sufficient cooling water to maintain the fuel cladding temperatures below temperature limits defined in 10CFR50.46.

In the event of a severe accident that results in a core melt with the molten core in the lower drywell region, GDCS will flood the lower drywell cavity region with the water inventory of the three GDCS pools and the suppression pool (SP).

The GDCS is an Engineered Safety Feature (ESF) system. It is classified as safety-related and Seismic Category I. GDCS instrumentation and dc power supply are IEEE Class 1E.

Basic system parameters are:

- Number of independent divisions: 3
- Initiation signal: confirmed Level 1 signal from NBS
  - Type: Sealed-in NBS divisional Level 1 signal
  - Number of channels: 4
- Time delay between initiation and actuation for short term water injection:
  - 150 seconds
- Time delay between initiation and actuation for long term water injection:
  - 30 minutes
  - Permissive: Interlocked to RPV water level ≤ (TAF + 1.0m)
- Squib valve firing logic: 2-out-of-3
- Manual actuation:
  - No. of channels: 4
  - Permissive: Interlocked to RPV low pressure signal
  - Logic: Simultaneous operation of two switches of the same division

The GDCS injects water into the downcomer annulus region of the reactor after a LOCA and reactor vessel depressurization. It provides short-term gravity-driven water makeup from three separate water pools located within the upper drywell at an elevation above the active core region. The system also provides long-term post-LOCA makeup from the suppression pool to meet long-term core decay heat boil-off requirements. During severe accidents the system floods the lower drywell region with water if the core melts through the RPV.

The GDCS is completely automatic in actuation and operation. A backup to automatic actuation is the ability to actuate by operator action.

The GDCS is composed of three identical divisions completely independent of each other both electrically and mechanically. A confirmed RPV Level 1 signal will actuate the ADS to reduce RPV pressure. Simultaneously, 150-second short-term system timers, and 30-minute long-term system timers in the GDCS logic are started, which, after time-out, actuate squib valves providing an open flow path from the respective water sources to the vessel.

The short-term system supplies gravity-driven flow to six separate nozzles on the vessel with suction flow from three separate GDCS pools. The long-term system supplies gravity-driven flow to three other nozzles with suction flow from the suppression pool through equalizing lines.

Both the short-term and long-term systems are designed to ensure that adequate reactor vessel inventory is provided assuming a LOCA in one division and failure of one squib valve to actuate in the second division.

Three GDCS deluge lines each having one squib actuated valve provide a means of flooding the lower drywell cavity in the event of a core melt sequence which causes failure of the lower vessel head and allows molten fuel to reach the lower drywell cavity floor. These squib activated valves are driven by logics receiving input signals from an array of temperature sensors located in the lower drywell.

### 1.2.2.5 Reactor Servicing Equipment

#### 1.2.2.5.1 Fuel Service Equipment

The refueling and fuel handling platform are also included and are outlined in Subsection 1.2.2.5.5. Servicing tools and equipment are not safety-related.

***Fuel Prep Machine***
One fuel prep machine is mounted against the west wall of the spent fuel storage pool. Its primary use is to inspect spent fuel when submerged in the storage pool and to aid in reconstitution of fuel found to be defective.

***New Fuel Inspection Stand***
The inspection stand is mounted in a pit next to the new fuel storage vault. The pit allows inspection of the two fuel bundles over their full length. Channeling is also performed with the aid of the channel handling tool.

***Channel Bolt Wrench***
A long handled socket-end wrench used in the assembly or disassembly of the channel from the fuel bundle, by insertion or removal of the attaching bolt, while channeling new fuel or reconstituting spent fuel.

***Channel Handling Tool***
A long handled clamping tool used to engage the channel for removal. It is manually operated and suspended from the auxiliary hoist or jib crane.

***Vacuum Sipper***
Used in the spent fuel pool to detect gases from defective fuel.

***General Purpose Grapple***
A general use grapple primarily for handling fuel when using any fuel handling equipment

### 1.2.2.5.2 Miscellaneous Service Equipment

This equipment is generally used independently of other servicing equipment. Equipment requirements are that they operate underwater to a depth of 33 meters. The equipment is designed to be quickly decontaminated and can be stored with a minimum of manpower.

***Underwater Lights***
Three types of lights are used; A general area light, a local area light, and a drop-type light.

***Viewing Aids***
Three types of viewing aids are used. A floating type viewing aid is the simplest. Another aid features an under water viewing tube with a 15-60 power telescope. The last is an underwater, remotely controlled television camera with an internal light source.

***Under Water Vacuum Cleaner***
The underwater vacuum cleaner is used to clean any pool floor underwater and is remotely serviceable while submerged.

### 1.2.2.5.3 Reactor Pressure Vessel Servicing Equipment

These tools are used when the reactor is shut down and the RPV head is being removed or installed. Lifting tools are designed for a safety factor of 10 or better with respect to the ultimate strength of the material used. Carbon steel equipment must be either hard chrome plated, parkerized or coated. Tools are designed for 60-year life in the working environment.

### General Tools

This group includes Stud Handling Tool, Stud Wrench, Nut Runner, Stud Thread Protector, Thread Protector Mandrel, Bushing Wrench, Seal Surface Protector, Stud Elongation Measuring Device, Dial Indicator Elongation Measuring Device, and Head Guide Cap.

### Steamline and DPV Nozzle Plug

The plugs are inserted into the main steam and DPV line nozzles prior to refueling when the reactor water is at the refueling level, to prevent water outflow during SRV, MSIV and DPV maintenance activities. The seals are housed at the end of a spider frame and when remotely inserted in to the nozzles are released from the plug frame. Each plug when actuated can seal against full head pressure.

The fixtures and spider frame area fabricated from corrosion resistant materials and are designed using a factor of safety of five or better.

### Chimney Head Stud Wrench

The wrench is hand held for loosening and tightening the chimney head studs. It is made of aluminum and is designed for a 60-year life.

### Head Support Pedestal

The reactor vessel head is supported by three equi-spaced pedestals mounted on the refueling floor. Each have dowel pins that engage the vessel flange stud holes at 0.9 m above the floor to afford access to the flange seal surface.

### Dryer Separator Strongback

The strongback is a cruciform-shaped structure used as a lifting device for the steam dryer/separator assembly. When lowered over the steam dryer by the main crane, four lifting pins are remotely actuated into engagement with the dryer lifting eyes.

### Head Strongback Tensioner

The head strongback forms the structural base on which the automatic stud tensioner is supported. The upper deck provides support for the tensioner power units and around the periphery is a rack and pinion drive by which the tensioners are moved in to their station over the stud centers. Below the strongback are lifting columns to which the lifting lugs on the vessel are engaged. The tensioner has the capability of retaining the vessel studs and nuts.

## 1.2.2.5.4 RPV Internals Servicing Equipment

### Instrument Strongback

The instrument strongback is used to aid in handling and replacement of power range neutron monitoring (PRNM) and startup range neutron monitoring (SRNM) dry tubes, in conjunction with support from the instrument handling tool.

***Instrument Handling Tool***

The instrument handling tool is connected to the wire terminal of the auxiliary hoist of the refueling platform and receives LPRMs or dry tubes from the strongback.

#### 1.2.2.5.5 Refueling Equipment

The reactor building fuel handling floor is serviced with a fuel handling platform, Refueling machine, and an auxiliary platform.

***Refueling Machine***

The refueling machine is a gantry-type crane which spans the reactor vessel cavity and fuel and storage pools to handle fuel and perform other ancillary tasks. It is equipped with a traversing trolley on which is mounted a telescoping tubular mast and integral fuel grapple. An auxiliary hoist is also provided. The machine is a rigid structure built to precise engineering standards to ensure accurate and repeatable positioning during the refueling process. A programmed computer located above the refueling floor controls the operational movements.

***Fuel Handling Platform***

Although similar in appearance and size to that of its counterpart, the fuel handling platform is only used for fuel servicing and transporting tasks. It is equipped with a trolley and telescoping grapple and is manually operated. Mechanical stops and interlocks provide the necessary operational limits.

***Auxiliary Platform***

The auxiliary platform is a low-profile structure having its own track located on the fuel handling area floor. A removable section with mounted wheels is lowered to the reactor vessel flange level on which a special portable track is installed. Its primary purpose is to aid in open vessel servicing.

#### 1.2.2.5.6 Fuel Storage Facility

New and spent fuel storage facilities are required for fuel and associated equipment. Storage in wet or dry conditions depends on the item in storage.

***New Fuel Storage***

New fuel storage racks are aluminum and are constructed for floor mounting. For dry vault storage the racks are loaded from the top, while those in pools are side loaded. The storage vault capacity is 19% of core load while the storage pool is 39% core load.

***Spent Fuel Storage***

Spent fuel storage racks are of stainless steel laminate construction with neutron absorbing material. This ensures that a full array (285% of full core) or loaded spent fuel will remain subcritical by 5% of $\Delta k$, under all conditions.

Adequate water shielding is always maintained in storage pools by the use of level sensors. Dry vaults on the other hand have drains to assure they are maintained dry. All storage pools are constructed with stainless steel liners to form a leak-tight barrier. A leak detection system monitors liner integrity.

The thermal-hydraulic design of the rack provides sufficient natural convection cooling flow to remove 19,929 W/bundle (68,000 Btu/hr/bundle) of decay heat.

### 1.2.2.5.7 Under-Vessel Servicing Equipment

The primary functions of the under vessel servicing equipment are to:

- install and remove fine motion control rod drives (FMCRD)
- install and remove FMCRD packing sections and motors
- make connections to Neutron Detectors
- provide servicing tools
- provide a work platform and CRD Handling Equipment

***Under-Vessel Platform***

The under-vessel platform provides a working surface for personnel and equipment to the entire under-vessel area. This requires 360° rotational capability. The platform also provides the facility for operation of the FMCRD handling machine for the automatic removal of the FMCRDs.

### 1.2.2.5.8 FMCRD Maintenance Area

The FMCRD maintenance area is designed and equipped to perform FMCRD maintenance related activities, including decontamination of the FMCRD components, acceptance testing, and storing spare drives. Maintenance tasks use a combination of manual and remote operations to reduce radiation exposure to plant personnel and to reduce contamination of surrounding equipment during operation.

The FMCRD maintenance area is located in a shielded room near the drywell equipment entry door. The layout of the room permits a convenient and efficient sequencing of work while reducing exposure to personnel.

### 1.2.2.5.9 Fuel Cask Cleaning

Spent fuel cask cleaning is performed in two different areas of the plant. Spent fuel cask cleaning is performed at the receiving area in the reactor building if required to remove surface dirt accumulated during transportation. It is also performed in the cask pit following loading of spent fuel, under the jurisdiction of health physics personnel.

The receiving area of the plant has facilities for:

- Checking the cask for contamination;
- Cleaning the cask of road dirt;
- Inspection of the cask for damage;
- Attachment of the cask lifting yoke;
- Removal of head bolts and attachment of head lifting cables; and
- Raising the cask to the refueling floor using the main building crane.

The cask pit area includes:

- A deep drainable pit with gate access to the storage pool for underwater cask loading.
- An underwater area for the storage of the cask head and lifting yoke.
- An area for high pressure cleaning and decontamination. This area is accessible for chemical and hand scrubbing, refastening the head, and for smear tests.

#### 1.2.2.5.10 Fuel Transfer System

The fuel is removed from the reactor and transported through the pool gate into the transfer pool where it is seated in the fuel basket of the transfer machine. The basket is then conveyed along the transfer pool west wall. The fuel handling platform will then grapple the fuel and place it in the spent fuel storage pool. From vessel removal to storage in the spent fuel pool, the fuel bundle is handled in the vertical position.

#### 1.2.2.5.11 Inservice Inspection Equipment

The SBWR typically uses a wide range of inservice inspection equipment much of which is equipment and materials used in performance of visual, surface and volumetric examinations required by the ASME Code, Section XI.

- Automated ultrasonic scanning equipment using multiple angle beam and straight beam transducers may be employed for volumetric examination of areas such as reactor pressure vessel welds and nozzle inner radii. The data from the automated examination is typically stored on optical disk or other appropriate recording media for subsequent computer-assisted data analysis.
- Manual ultrasonic examination equipment may be employed to supplement the automated examination if necessary or to perform the volumetric examination of

areas such as ASME Class 2 vessel welds and nozzle inner radii. Manual ultrasonic examination equipment consists of an ultrasonic instrument containing analog or digital oscilloscope-style display and hand-held transducers. Where more than one angle beam examination is required due to the Class 2 vessel wall thickness, additional manual scans may be performed using ultrasonic transducers adjusted for the required angles of examination. Class 1 and 2 piping welds may be examined volumetrically using either computerized, automated ultrasonic scanning equipment or using manual ultrasonic examination equipment.

- Surface examinations of ferritic vessels and piping may be performed using the magnetic particle examination method with either prod or yoke type equipment. The magnetic particles may be either dry or may be in a wet suspension and may be either fluorescent or colored for viewing in visible light. Surface examinations of non-magnetic vessel and piping welds may be performed using either fluorescent or visible dye liquid penetrant materials. When fluorescent magnetic particles or liquid penetrant materials are used, portable ultraviolet lights are used for viewing.

- Eddy-current probe coils driven by automated scanning devices with computerized data acquisition systems may be substituted for surface examinations where the component configuration or radiation conditions render other surface examination techniques impractical or undesirable.

- Visual examinations of Class 1 and 2 bolting and component supports and attachments on Class 1, 2 and 3 piping and components may be conducted directly using simple aids such as mirrors and magnifying glasses.

- Remote visual examination equipment may be used for examination of interior surfaces of the reactor vessel and other components. Rigid fixtures are sometimes used as an aid in performance of the remote reactor visual examinations.

It is anticipated there will be continuing technological advances in inservice inspection. As these improved technologies become available and proven, they will be applied (as appropriate) to inspection of the certified design.

#### 1.2.2.6 Reactor Auxiliary Systems

##### 1.2.2.6.1 Reactor Water Cleanup/Shutdown Cooling System

The Reactor Water Cleanup/Shutdown Cooling (RWCU/SDC) System has the following primary functions:

- Purifies reactor coolant during normal operation and shutdown.

- Transfers sensible and core decay heat produced when the reactor is being shutdown or is in the shutdown condition.

- Provides decay heat removal and high pressure cooling of the primary coolant during periods of reactor isolation (hot standby)
- Implements the overboarding of excess reactor coolant during startup and hot standby.
- Maintains coolant flow from the reactor vessel bottom head to reduce thermal stratification.
- Warms the reactor coolant prior to startup and hydrotesting.

The system consists of two redundant trains. Each train includes a pump powered and controlled by an adjustable speed drive (ASD), two regenerative heat exchangers (RHX), one single-shell non-regenerative heat exchanger (NRHX), one radial-bed-type low-pressure-drop resin-bed demineralizer, an electric heater, and associated valves and pipes.

The RWCU/SDC System is classified as a non-safety-related system except for its RCPB and containment isolation functions which are safety-related and is thus Seismic Category I and Class 1E.The electrical power supplies to the two trains are from separate electrical divisions. The system can be connected to non-safety-related standby ac power (diesel generators).

During normal plant operation, the system operates at reduced flow in the cleanup mode continuously withdrawing water from RPV. The water is cooled through the heat exchangers and is circulated by the pump to the demineralizer for removal of impurities. Purified water returns to the RHX where it is reheated, and then flows into the feedwater lines and is returned to the RPV. One train is in operation while the other is in standby.

Redundant trains permit shutdown cooling if only one train is operable. The cooldown time will be extended when using only one train. In the event of loss of preferred power and the most limiting single active failure, this mode of operation brings the RPV to a ≤100°C (≤212°F) cold shutdown condition in 36 hours in conjunction with operation of the Isolation Condensers. The RWCU/SDC provides the shutdown cooling capability to satisfy the following reactor coolant temperature reduction schedule:

- 60°C (140°F) in 24 hours
- 54.4°C (130°F) in 40 hours
- 48.9°C (120°F) at the completion of flooding the reactor well from ≤35°C (≤95°F) water sources

During hot standby and startup, excess water resulting from CRD system purge water injection and expansion during plant heatup is dumped, or overboarded, to the main condenser or the radwaste system to control reactor water level.

The RWCU/SDC System maintains the temperature difference between the reactor dome and the bottom head drain to less than 80.6°C (145°F) to preclude excessive thermal stratification.

Flow rate, pressure, temperature and conductivity are measured, recorded or indicated, and alarmed if appropriate, in the MCR.

Pumps are provided with interlocks for the automatic operation and with switch and status indication for manual operation from the MCR. Motor operated isolation valves are automatically and manually actuated with automatic closure overriding manual opening signals.

#### 1.2.2.6.2 Fuel and Auxiliary Pools Cooling System

The FAPCS performs pool water cooling, purification, and distribution (i.e., pool filling and, where applicable, draining) for the following pools in the Reactor Building:

- Spent fuel storage pool;
- Fuel cask pit;
- Fuel transfer pool;
- New fuel storage pool;
- Reactor well;
- Skimmer surge tank;
- Isolation condenser (IC/PCC) pools;
- Gravity-Driven Cooling System (GDCS) (3 pools); and
- Suppression pool.

In addition, the FAPCS performs the following system functions:

- Reactor well draining and filling;
- Drywell spray;
- Suppression chamber spray; and

- Low Pressure Coolant Injection (LPCI) of suppression pool water into the RPV.

The FAPCS is a low gauge pressure system which has two trains of components. Each train includes a pump, a heat exchanger and a filter-and-demineralizer water treatment unit. At each end, the trains are tied together by a four-valve bridge of motor-operated valves, which are aligned to perform the system functions. The FAPCS has features to prevent radioactive contamination of the IC/PCC pool with untreated water from other pools.

One train of the FAPCS normally operates continuously to cool, clean and clarify the water of the spent fuel storage pool. The other train is in standby or may be performing periodic cooling/cleaning of one of the other pools. FAPCS provides sufficient flowrate and cooling capability to keep the spent fuel pool bulk water temperature at or below 48.9°C (120°F) for normal plant operations and normal spent fuel pool heat load conditions. With conditions associated with a full core off-load and irradiated fuel in the spent fuel pool for 10 years of plant operations, the FAPCS maintains the bulk temperature at or below 60°C (140°F). The same capability remains if there is a single failure in the FAPCS.

FAPCS operation is manually controlled and monitored from the MCR.

The water treatment is locally controlled with status feedback to the MCR. Automatic operation applies to sequential logic for start/stop of pumps and line-up of the valves to assure the selection of cooling/clean-up train configuration and system operating modes. The operator is able to override the automatic control and take manual control. Containment isolation operation has priority over normal operation of the system and is controlled by the LD&IS.

The system contains instrumentation for sensing and transmitting water levels, water temperatures, water flow and water pressure.

The FAPCS is a non-safety-related system with the exception of containment isolation and the independent safety-related makeup water piping providing makeup water to the isolation condenser (IC/PCC) pools and independently to the spent fuel pool. The piping and components directly interacting with safety-related systems meet the classification of safety and seismic class required by these other systems.

### 1.2.2.7 Control Panels

#### 1.2.2.7.1 Main Control Room Panels

The main control room panel is comprised of an integrated set of operater interface panels (e.g., main control console, large display panel). The safety-related panels are seismically qualified and provide grounding, electrical independence and physical separation between safety divisions and between safety divisions and non-safety-related components and wiring.

The main control room panels and other MCR operator interfaces are designed to provide the operator with information and controls needed to safely operate the plant in all operating modes, including startup, refueling, safe shutdown, and maintaining the plant in a safe shutdown condition. Human factors engineering principles have been incorporated into all aspects of the SBWR MCR design.

#### 1.2.2.7.2 Radwaste Control Room Panels

The liquid and solid radwaste systems are operated from control panels in the radwaste control room. Programmable controllers are used in this application. They are not safety-related.

#### 1.2.2.7.3 Local Control Panels and Racks

Local panels, control boxes, and instrument racks are provided as protective housings and/or support structures for electrical and electronic equipment to facilitate system operations at the local level. They are designed for uniformity using rigid steel structures capable of maintaining structural integrity as required under seismic and plant dynamic conditions. The term "local panels" includes local control boxes.

Local panels and racks used for plant protection systems are classified as safety-related. They are located in areas in which there are no potential sources of missiles or pipe breaks that could jeopardize modules from more than one division. Each safety-related panel/rack is Seismic Category I, qualified, and provides grounding, and electrical independence and physical separation between safety divisions and non-essential components and wiring.

Electrical power to divisional panels/racks is from ac or dc power sources of the same division as that of each panel/rack itself. Power to the non-essential panels/racks is from the non-essential ac and/or dc sources.

#### 1.2.2.7.4 Essential Multiplexing Subsystem

The essential multiplexing Subsystem (EMS) provides distributed data acquisition and control networks to support the monitoring and control of the plant standby safety systems. EMS comprises electrical devices and circuitry, such as local multiplexing units (LMUs), fiber optic transmission lines, and control room multiplexing units (CMUs), that acquire data from remote process sensors and discrete monitors located within the plant and multiplex the signals to SSLC equipment. SSLC provides decision logic that trips the final actuators of driven equipment associated with safety systems.

EMS is divided into four divisions of equipment, each with independent control of data acquisition, multiplexing, and control output functions. System timing is asynchronous among the four divisions. No common clock signal is transmitted among the divisions of multiplexing and no timing signals are exchanged.

Both analog and discrete sensors are connected to LMUs in local areas, which perform signal conditioning, analog-to-digital conversion for continuous process inputs, change-of-state detection for discrete inputs, and message formatting prior to signal transmission. The LMUs are limited to acquisition of sensor data and the output of control signals. Trip decisions and other control logic functions are performed in SSLC processors. The LMUs transmit serial, time-multiplexed data streams representing the status of the plant variables to the SSLC logic processing equipment. Data transmission is also made over dual redundant channels to the main control room. The CMUs demultiplex the data and prepare the signals for use in interfacing monitoring systems such as the process computer or display controllers. The CMUs also receive safety-related signals from control room equipment for transmission to the LMUs and SSLC. EMS design features automatic self-test and automatic reconfiguration after failure of one channel (either a cable break or device failure). If an LMU or CMU has failed, that unit will be removed from service. Faults and their location are annunciated to the operator in the MCR.

Data can be transferred to non-safety-related systems for control or display through isolating fiber-optic data links and buffering devices (gateways or bridges, if required). Data transfer is made such that failures on the non-safety side cannot inhibit operation of safety-related logic functions. Data cannot be transmitted from the non-safety side to EMS.

EMS is capable of data transfer at rates sufficient to satisfy the system time response requirements of safety system functions. Data throughput capability is up to 100 megabits per second.

EMS starts and runs automatically upon application of system power, regardless of the sequence in which power is applied to individual controllers. EMS and SSLC automatically establish communications by detection of correct message passing. Logic is provided to prevent equipment activation outputs from occurring until stable plant sensor data and interlock permissive data are being received.

Loss of power causes a controlled transition to a safe-state without transients occurring that could cause inadvertent initiation or shutdown of driven equipment.

EMS equipment is classified as safety-related, Class 1E, and is Seismic Category I.

EMS includes test facilities in the MCR that will monitor data transmission to ensure that data transport, routing, and timing specifications are accurate. Bit error rate of each EMS network shall be better than 1 error in $10^9$. Out-of-tolerance parameters detected on-line for a particular input signal will result in an inoperative condition for that input into the trip logic processors of SSLC.

#### 1.2.2.7.5 Non-Essential Multiplexing System

The Non-Essential Multiplexing System (NEMS) is the data communication portion of all control systems in the plant that are not part of the shutdown control systems. The NEMS in non-safety-related.

The NEMS equipment is designed and constructed using state-of-the-art fiber optics communications equipment and computer controls which perform the following:

- Transfer via the NEMS to control system equipment, in digital format, analog or binary data that has been collected and digitized from remote transmitters, contact closures, and other sensors located throughout the plant.
- Transfer from the main control room via the NEMS to control system equipment, in digital format, processed activation signals for the control of remote devices such as pumps, valves, or solenoids.
- Exchange self-test data between local equipment and control room equipment for the reporting of NEMS and control system component malfunctions.
- Communicate requests to the main control room for the reporting of NEMS and control system equipment component malfunctions.

The NEMS has no access to the safety-related data base; however safety-related data can be read by the NEMS on the optically-isolated memory portions of the Essential Multiplexing System (EMS) Local Multiplexing Units (LMU). This data can be read by any NEMS multiplexing units that is configured to do so. The NEMS cannot write data to any portion of the EMS.

The NEMS consists of two types of multiplexing units: Local Multiplexing Units (LMU), and Control Room Multiplexing Units (CMU) connected via fiber optic cables. The NEMS also includes network gateways which allow transfer of data between data highway systems.

Throughout the plant, LMUs are located in local plant areas to acquire sensor data and transmit this data to the any equipment that requires it. The LMUs also receive processed signals from the control room for command of control system actuators. CMUs are located in the control room to transmit and receive data for the logic processing units of the plant control systems.

All interconnections are fiber optic data links. Within each NEMS highway, the system uses redundant links for greater reliability.

There are a number of NEMS highway systems that are routed throughout the plant. These systems all have CMUs located in the main control room. Gateways connect the multiple NEMS highway systems to allow for transfer of data between NEMS highway systems.

### 1.2.2.8 Nuclear Fuel

#### 1.2.2.8.1 Nuclear Fuel

Fuel design for the SBWR Standard Plant is not within the scope of the certified design. It is intended that the specific fuel to be used in any facility which has adopted the certified design be in compliance with U.S. NRC approved fuel design criteria. This strategy is intended to permit future use of enhanced/improved fuel designs as they become available. However, this approach is predicated on the assumption that future fuel designs will be extensions of the basic fuel technology that has been developed for boiling water reactors. Key characteristics of this established BWR fuel technology are:

- Uranium oxide based fuel pellets;
- Zirconium-based (or equivalent) fuel cladding;
- All material selected on the basis of BWR operating conditions;
- Multi-rod fuel bundles in an N lattice; and
- Fuel bundle inlet orificing to control bundle flow rates, core flow distribution, and reactor coolant hydraulic characteristics.

The SBWR design provides a Loose Parts Monitoring System (LPMS) aimed at protecting the fuel against the potential effects of loose parts entrained in the reactor coolant flow. A discussion of the LPMS is included in this section.

The following is a summary of the principal requirements which must be met by the fuel supplied to any facility utilizing the certified design.

***General Criteria***

- NRC-approved analytical models and analysis procedures are applied.
- New design features are included in lead test assemblies.
- The generic post-irradiation fuel examination program approved by NRC is maintained.

### Thermal-Mechanical

The fuel design thermal-mechanical analyses are performed for the following conditions:

- Either worst tolerance assumptions are applied or probabilistic analyses are performed to determine statistically bounding results (i.e., upper 95% confidence).
- Operating conditions are taken to bound the conditions anticipated during normal steady-state operation and anticipated operational occurrences.

The fuel design evaluations are performed against the following criteria:

- The fuel rod and fuel assembly component stresses, strains, and fatigue life usage are evaluated to not exceed the material ultimate stress or strain and the thermal fatigue capability.
- Mechanical testing is performed to ensure that loss of fuel rod and assembly component mechanical integrity will not occur due to fretting wear.
- The fuel rod and assembly component evaluations include consideration of metal thinning and any associated temperature increase due to oxidation and the buildup of corrosion products to the extent that these influence the material properties and structural strength of the components.
- The fuel rod internal hydrogen content is controlled during manufacture of the fuel rod consistent with ASTM standards.
- The fuel rod is evaluated to ensure that fuel rod bowing does not result in loss of fuel rod mechanical integrity due to boiling transition.
- Loss of fuel rod mechanical integrity will not occur due to excessive cladding pressure loading.
- The fuel assembly (including channel box), control rod and CRD are evaluated to assure control rods can be inserted when required. These evaluations consider the effect of combined safe shutdown earthquake (SSE) and LOCA loads.
- Loss of fuel rod mechanical integrity will not occur due to cladding collapse into a fuel column axial gap.
- Loss of fuel rod mechanical integrity will not occur due to pellet-cladding mechanical interaction.

### Nuclear

- A negative Doppler reactivity coefficient is maintained for any operating condition.

- A negative core moderator void reactivity coefficient resulting from boiling in the active flow channels is maintained for any operating conditions.
- A negative moderator temperature reactivity coefficient is maintained above hot standby.
- For a super prompt critical reactivity insertion accident originating from any operating condition, the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative.
- A negative power reactivity coefficient, as determined by calculating the reactivity change, due to an incremental power change from a steady-state base power level, is maintained for all operating power levels above hot standby.
- The plant meets the cold shutdown margin requirement.
- The effective multiplication factor for fuel designs stored under normal and abnormal conditions is shown to meet fuel storage limits by demonstrating that the peak uncontrolled lattice k-infinity calculated in a normal reactor core configurations meets the limits for the storage racks.

### Hydraulic

Flow pressure drop characteristics are included in the calculation of the operating limit minimum critical power ratio (MCPR).

Because of the channeled configuration of BWR fuel assemblies, there is no bundle-to-bundle cross-flow inside the core, and the only issue of hydraulic compatibility of various bundle types in a core is the bundle inlet flow rate variation and its impact on margin-to-thermal limits. The coupled thermal-hydraulic-nuclear analyses performed to determine fuel bundle flow and power distribution uses the various bundle pressure loss coefficients to determine the flow distribution required to maintain a total core pressure drop boundary condition to be applied to all fuel bundles. The margin to the thermal limits of each fuel bundle is determined using this consistent set of calculated bundle flow and power.

### Loose Parts Monitoring System

The Loose Parts Monitoring System (LPMS) is designed to provide detection of loose metallic parts within the RPV. Detection of loose parts can provide the time required to avoid or mitigate safety-related damage to or malfunctions of primary system components. The LPMS detects structure borne sound that can indicate the presence of loose parts impacting against the RPV internals. The system alarms when the signal amplitude exceeds preset limits. The LPMS detection system can evaluate some aspects of selected signals. However, the system by itself will not diagnose the presence and location of a loose part. Review of LPMS data by an experienced LPM engineer is required to confirm the presence of a loose part.

The LPMS continuously monitors the RPV and appurtenances for indications of loose parts. The LPMS consists of sensors, cables, signal conditioning equipment, alarming monitor, signal analysis and data acquisition equipment, and calibration equipment. The alarm setting is set low enough to meet the sensitivity requirements, yet is designed to discriminate between normal background noises and the loose part impact signal to minimize spurious alarms.

The array of LPMS sensors consist of a set of sensor channels that are strategically mounted on the external surface of the primary pressure boundary at various elevations and azimuths at natural collection regions for potential loose parts. General mounting locations are at the a) main steam outlet nozzle, b) feedwater inlet nozzle, and, (c) control rod drive housings.

The online system sensitivity is such that the system can detect a metallic loose part that weighs between 0.25 lb to 30 lbs and impacts with a kinetic energy of 0.5 ft.-lb on the inside surface of the RPV within 3 feet of a sensor. The LPMS frequency range of interest is typically from 1 to 10 kHz. Frequencies lower than 1 kHz are generally associated with flow induced vibration signals or flow noise.

The LPMS includes provisions for both automatic and manual start-up of data acquisition equipment with automatic activation in the event the preset alert level is reached or exceeded. The system also initiates an alarm to the control room personnel when an alert condition is reached.

#### 1.2.2.8.2 Fuel Channel

Fuel channel design for the SBWR is not within scope of the certified design. It is intended that the specific fuel channel to be used in any facility which has adopted the certified design be in compliance with U.S. NRC approved fuel channel design criteria. This strategy is intended to permit future use of enhanced/improved fuel channel designs as they become available. However, this approach is predicated on the assumption that future fuel channel designs will be extensions of the basic technology that has been developed for boiling water reactors. The key characteristic of this established BWR fuel channel technology is the use of zirconium-based (or equivalent) fuel channels which preclude cross-flow in the core region.

The following is a summary of the principal requirements which must be met by the fuel channel supplied to any facility using the certified design:

- The material of the fuel channel shall be shown to be compatible with the reactor environment.

- The channel will be evaluated to ensure that channel deflection does not preclude control rod drive operation.

- The effects of channel bow will be included in the fuel rod critical power evaluations.

#### 1.2.2.8.3 Control Rod

Control rod design for the SBWR is not within the scope of the certified design. It is intended [illegible]t the specific control rod to be used in any facility which has adopted the certified [illegible]gn be in compliance with U.S. NRC approved control rod design criteria. [illegible]his stra[illegible] is intended to permit future use of enhanced/improved control rod designs as [illegible]ey become available. However, this approach is predicated on the assumption that future control rod designs will be extensions of the basic technology that has been developed for boiling water reactors. Key characteristics of this established BWR control rod technology are:

- Control rods perform dual functions of power distribution shaping and reactivity control.

- The control rod has a cruciform cross-sectional envelope shape.

- The control rod has a coupling at the bottom for attachment to the CRD.

- The control rod has an upper bail handle for transporting.

- The cruciform cross section contains neutron poison materials which are either contained within or as part of the control rod structure.

The following is a summary of the principal requirements which must be met by the control rod supplied to any facility utilizing the certified design:

- The control rod stresses, strains, and cumulative fatigue shall be evaluated to not exceed the ultimate stress or strain of the material.

- The control rod shall be evaluated to be capable of insertion into the core during all modes of plant operation within the limits assumed in the plant analyses.

- The material of the control rod shall be shown to be compatible with the reactor environment.

- The reactivity worth of the control rod shall be included in the plant core analyses.

- Lead Surveillance program shall be implemented if a change in design features such as new absorber material or structural material not previously used in reactor cores could impact the function of the control rod.

### 1.2.2.9 Radioactive Waste Management System

#### 1.2.2.9.1 Liquid Waste Management System

The Liquid Waste Management System (LWMS) collects, monitors, and treats liquid radioactive waste for plant reuse whenever practicable.

The LWMS consists of the following six subsystems:

- Equipment (low conductivity) drain subsystem;
- Floor (high conductivity) drain subsystem;
- Chemical drain subsystem;
- Detergent drain subsystem;
- Mobile systems interface subsystem; and
- Mixed waste subsystem.

The LWMS processing equipment is located in the radwaste building. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accord with applicable local, state, and federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant. These wastes are transferred to collection tanks in the radwaste building.

Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of suspended solids and chemical contaminants. Equipment drains and other low-conductivity wastes are treated by filtration, uv/ozone, demineralization and are transferred to tne condensate storage tank for reuse. Floor drains and other high conductivity wastes are treated by filtration and ion exchange prior to being either discharged or recycled for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway. Chemical wastes are treated by filtration, sampled and released from the plant on a batch basis. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls. Connections are provided for mobile processing systems that could be brought in to augment the installed waste processing capability.

Connections for addition of a permanent evaporation subsystem are provided in the event that site conditions warrant. Mixed waste will be segregated from the other types of radioactive waste for packaging.

If the liquid is returned to the plant, it meets the purity requirements for condensate makeup. If the liquid is discharged, the activity concentration is consistent with the discharge criteria of 10CFR20 and dose commitment in 10CFR50, Appendix I.

#### 1.2.2.9.2 Solid Waste Management System

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and temporarily store prior to shipment solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences, that includes filter backwash sludges and bead resins generated by the LWMS, RWCU/SDC, FAPCS, and condensate system. Contaminated solids such as High Efficiency Particulate Air and cartridge filters, rags, plastic, paper, clothing, tools, and equipment are also processed in the SWMS. There is no liquid plant discharge from the SWMS.

The SWMS consists of the following four subsystems:

- Wet solid waste collection subsystem;
- Wet solid waste processing subsystem;
- Dry solid waste processing subsystem: and
- Mobile systems interface subsystem.

Spent bead resin sluiced from the RWCU/SDC System, FAPCS, condensate and LWMS are transferred by the wet solid waste collection subsystem to one of two spent resin tanks for decay and storage.

The wet solid waste processing subsystem consists of a built-in dewatering station. A High Integrity Container (HIC) is filled with either sludges from the phase separator or bead resin from the spent resin tanks. Spent cartridge filters may also be placed in the HIC.

Dry wastes consist of air filters, miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; and solid laboratory wastes. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant. The filled containers are sealed and moved to controlled-access enclosed area for temporary storage.

Connections are provided for mobile processing systems that could be brought in to augment the installed waste processing capability.

Connections for addition of a permanent solidification subsystem are provided in the event that site conditions warrant.

Temporary storage for over one month's volume of packaged waste is provided in the radwaste building. Packaged waste includes high integrity containers, compactor boxes, shielded filter containers, and 55-gallon drums as necessary.

The SWMS is designed to package the radioactive solid waste for off-site shipment and burial, in accordance with the requirements of applicable NRC and DOT regulations, including Regulatory Guide 1.143, 10CFR61, 10CFR71, and 49CFR170 through 178.

#### 1.2.2.9.3 Gaseous Waste Management System

The function of gaseous waste management system is to minimize and control the release of radioactive material into the atmosphere by delaying, filtering, or diluting various offgas process and leakage gaseous releases which may contain the radioactive isotopes of krypton, xenon, iodine, and nitrogen. The Offgas System (OGS) is the principal gaseous waste management subsystem. The various building HVAC systems perform other gaseous waste functions.

The OGS provides for holdup and decay of radioactive gases in the offgas from the steam jet air ejector (SJAE) and consists of process equipment along with monitoring instrumentation and control components.

The OGS design minimizes the explosion potential in the offgas process stream through recombination of radiolytic hydrogen and oxygen under controlled conditions. Although the OGS is non-safety-related, it is capable of withstanding an internal hydrogen explosion and is designed to ASME Code Section VIII-Division I and the ANSI B31.1 Piping Code.

The OGS includes redundant hydrogen/oxygen catalytic recombiners and ambient temperature charcoal beds to provide for process gas volume reduction and radionuclide retention/decay. The system processes the SJAE discharge during plant startup and normal operation before discharging the air flow to the plant stack.

The charcoal beds can operate in three different modes:

- Bypass — all flow bypasses the beds (used during startup);
- Guard bed — all flow passes through the guard bed only; and
- Adsorber beds — all flow passes through the guard bed and then through parallel pairs of adsorber beds.

### 1.2.2.10 Power Cycle

#### 1.2.2.10.1 Turbine Main Steam System

The Turbine Main Steam (TMSS) System conveys steam generated in the reactor to the turbine. It also provides steam to the steam jet air ejectors, the turbine gland seals, the deaeration section of the main condenser, and the turbine bypass system. System boundaries are from after the outermost containment isolation valves up to the turbine stop valves.

The TMSS is not required to effect or support safe shutdown of the reactor or to perform in the operation of reactor safety features; however, the MS System is designed:

- to comply with applicable codes and standards to accommodate operational stresses such as internal pressure and dynamic loads without risk of failures and consequential releases of radioactivity in excess of the established regulatory limits;
- to accommodate nonnal and abnormal environmental limits;
- to assure that failures of non-Seismic Category I equipment or structures, or pipe cracks or breaks in high or moderate piping in the MS will not preclude functioning of safety-related equipment or structures in the plant; and
- with access to permit inservice testing and inspections.

The TMSS main steam piping consists of two lines. The header arrangement upstream of the turbine stop valves allows them to be tested on-line with minimum load reduction and also supplies steam to the power cycle auxiliaries, as required.

#### 1.2.2.10.2 Condensate and Feedwater System

The Condensate and Feedwater System (C&FS) consists of the piping, valves, pumps, heat exchangers, controls and instrumentation and the associated equipment and subsystems which supply the reactor with heated feedwater in a closed steam cycle utilizing regenerative feedwater heating. The C&FS extends from the main condenser outlet to the second feedwater isolation valve outside of containment.

The C&FS provides a dependable supply of high quality feedwater to the reactor at the required flow, pressure and temperature. The condensate pumps take the deaerated condensate from the condenser hotwell and deliver it through the SJAE condenser, the gland steam condenser, the condensate demineralizer and through a string of four low pressure feedwater heaters to the reactor feed pump suction. The reactor feed pumps discharge through two high pressure feedwater heaters to the reactor. Turbine extraction steam is used for a total of six stages of closed feedwater heating. The drains from each stage of the feedwater heaters are cascaded through successively lower pressure feedwater heaters to the main condenser.

The condensate portion of the C&FS has three motor-driven, constant speed centrifugal pumps, each rated at 33% to 60% of total system-rated flow.

The feedwater portion of the C&FS has three pumps operating in parallel, each rated at 33% to 60% of total system rated flow with adjustable speed motor drives with variable frequency power supplies.

The C&FS does not serve or support any safety function and has no safety design basis. Failure of this system cannot compromise any safety-related systems or prevent safe shutdown.

Portions of the system that are radioactive during operation are shielded with access control for inspections.

Leakage is minimized with welded construction used wherever practicable.

Relief discharges and operating vents are channeled through closed systems.

The majority of the C&FS piping is located within the turbine building which contains no safety-related equipment or systems. The portion which connects to the second isolation valve outside the containment is located in the steam tunnel in the reactor building. This portion of the piping is analyzed for dynamic effects from postulated events and SRV discharges.

The entire system piping is analyzed for waterhammer loads that could potentially result from anticipated flow transients.

#### 1.2.2.10.3 Condensate Purification System

The Condensate Purification System (CPS) continuously purifies and treats the condensate as required to maintain reactor feedwater purity, using filtration to remove solid corrosion products, ion exchange to remove condenser leakage and other dissolved impurities, and water treatment additions to minimize corrosion/erosion product releases in the power cycle.

The CPS does not serve or support any safety function and has no safety design basis. It is designed to Quality Group D standards.

Vent gases and other wastes from the CPS are collected in controlled areas and sent to the radwaste system for treatment and/or disposal.

The CPS is located in the turbine building, and piping or equipment failures will not affect plant safety.

### 1.2.2.10.4 Main Turbine

The main turbine is a tandem compound, two flow, 52 inch last stage bucket with one high pressure (HP) turbine and one low pressure (LP) turbine. The steam passes through an in-line high velocity moisture separator (HVS) prior to entering the LP turbine. Steam exhausted from the LP turbine is condensed and degassed in the condenser. The turbine uses steam at an atmospheric pressure of 6.79 MPa (985 psia) from the reactor and rotates at 1800 RPM. Steam is bled off from each turbine and is used to heat the feedwater. The steam and power conversion system is designed to operate at 105% of maximum guaranteed turbine throttle flow for transients and short-term loading conditions.

***Turbine Overspeed Protection System***

In addition to the normal speed control function provided by the turbine control system, a separate turbine overspeed protection system is included to minimize the possibility of turbine failure and high energy missile damage.

The following component redundancies are employed to guard against overspeed:

- Main stop valves/control valves;
- Intermediate stop valves/intercept valves (CIVs);
- Primary speed control/backup speed control;
- Fast acting solenoid valves/emergency trip fluid system (ETS); and
- Speed control/overspeed trip/backup overspeed trip.

The TG System is enclosed within the turbine building, which contains no safety-related equipment or structures. The turbine generator is orientated within the turbine building to be inline with the reactor building to minimize the potential for any high energy TG System generated missiles from damaging any safety-related equipment or structures.

### 1.2.2.10.5 Turbine Gland Seal System

The Turbine Gland Seal System (TGSS) provides steam and prevents the escape of radioactive steam from the turbine shaft/casing penetrations and valve stems and prevents air in-leakage through subatmospheric turbine glands.

The TGSS consists of a sealing steam pressure regulator, sealing steam header, a gland steam condenser, two full capacity exhaust blowers and associated piping, valves and instrumentation.

The TGSS is non-safety-related system and is designed to Quality Group D standards

The HP turbine shaft seals must accommodate a range of turbine shell pressure from full vacuum to approximately 220 psia. The LP turbine shaft seals operate against a vacuum at all times. The gland seal outer portion steam air mixture is exhausted to the gland steam condenser via the seal vent annulus (i.e. end glands), which is maintained at a slight vacuum. The radioactive content of the sealing steam, which eventually exhausts to the plant vent and the atmosphere, makes a negligible contribution to overall plant radiation release. In addition, the auxiliary steam system is designed to provide a 100% backup to the normal gland seal process steam supply. A full capacity gland steam condenser is provided and equipped with two 100% capacity blowers.

The TGSS effluents are first monitored by a system dedicated continuous radiation monitor installed on the gland steam condenser exhauster blower discharge. High monitor readings are alarmed in the MCR.

#### 1.2.2.10.6 Turbine Bypass System

A Turbine Bypass System (TBS) is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The TBS has the capability to shed 40% of the turbine generator rated load without reactor trip or operation of a SRV. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation.

The TBS does not serve or support any safety-related function and has no safety design.

Both automatic and manual control of the turbine bypass valves are provided. The turbine bypass valves are opened by a signal received from the SB&PC System whenever the actual steam pressure exceeds the preset steam pressure by a small margin. This occurs when the amount of steam generated by the reactor cannot be entirely used by the turbine. This bypass demand signal opens the first of the individual valves. As the bypass demand increases, additional bypass valves are opened, dumping the steam to the condenser. Pressure-reducing orifices are located at the condenser connections, and sparger piping distributes the steam within the condenser. The bypass valves are equipped with fast-acting servo valves to allow rapid opening of bypass valves upon turbine trip or generator load rejection.

The bypass valves automatically trip open upon load rejection or turbine trip. The bypass valves automatically trip closed whenever the vacuum in the main condenser falls below a preset value and/or insufficient circulating water flow exists. The bypass valves also fail closed on loss of electrical power or hydraulic system pressure.

### 1.2.2.10.7 Main Condenser

The main condenser is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the Turbine Bypass (TB) System.

The main condenser does not serve or support any safety function and has no safety design basis. It is, however, designed with necessary shielding and controlled access to protect plant personnel from radiation.

The main condenser is a single-shell type deaerating unit with this shell located directly beneath the low pressure turbine. The shell has tube bundles through which circulating water flows. The condensing steam is collected in the condenser hotwells (the lower shell portion) which provide suction to the condensate pumps.

Since the main condenser operates at a vacuum, any leakage is into the shell side of the main condenser. Tubeside or circulating water inleakage is detected by measuring the conductivity of sample water extracted beneath the tube bundles. In addition, conductivity is continuously monitored at the discharge of the condensate pumps and alarms are provided in the MCR.

In all operational modes, the condenser is at vacuum and consequently no radioactive releases can occur. Loss of vacuum sequentially leads to a control room alarm, turbine trip and eventually bypass and MSIV closure to prevent condenser overpressurization.

Ultimate overprotection is provided by rupture diaphragms on the turbine exhaust hoods.

The instrumentation and control features that monitor the performance to ensure that the condenser is in the correct operating mode include:

- Hotwell water level — Automatically controlled within preset limits. During normal full load operation with nominal hotwell levels, the main condenser provides a four-minute active condensate storage volume and has a two-minute surge capacity. At minimum normal operating hotwell water level, and normal full load condensate flow rate, the condenser provides a two minute minimum holdup time for N-16 decay.

- Condenser pressure — Key overall performance indicator that initiates alarms and trips at preset levels.

- LP turbine exhaust hood temperature — Automatically initiates turbine exhaust water sprays to protect the turbine.

- Inlet and outlet circulating water temperature — Monitors performance only.

- Conductivity within the condenser and at the discharge of the condensate pumps — Initiates alarms at preset levels.

The potential for flooding from the main condenser is less than that from the Circulating Water System (CWS) so only the CWS flooding protection is needed. The Condenser pressure indicators are located above any potential flood level.

Spray pipes and baffles are designed to protect the main condenser internals from high energy flow inputs.

Hydrogen buildup during operation is prevented by continuous evacuation of the main condenser. Hydrogen sources are excluded during shutdown.

Noncondensable gases are removed from the power cycle by the Main Condenser Evacuation System (MCES). The MCES removes power cycle noncondensable gases including the hydrogen and oxygen produced by radiolysis of water in the reactor and exhausts them to the OGS during plant power operation, or to the turbine building ventilation system exhaust during early plant startup. The MCES establishes and maintains a vacuum in the condenser by the use of steam jet air ejectors during power operation, and by a mechanical vacuum pump during early startup.

The system consists of two 100% capacity, double stage, steam jet air ejector (SJAE) units complete with intercondenser, for power plant operation, and a mechanical vacuum pump for use during startup. The last stage of the SJAE is a noncondensing stage. One SJAE unit is normally in operation and the other is on standby.

The steam jet air ejector is placed in service to remove the gases from the main condenser after a pressure of about 11 to 22 cm Hg (5 to 10 inches Hg) absolute is established in the main condenser by the mechanical vacuum pump and when sufficient nuclear steam pressure is available. Steam supply to the second stage ejector is maintained at a minimum specified flow to ensure adequate dilution of the hydrogen to prevent the offgas from reaching the flammable limit of hydrogen.

### 1.2.2.10.8 Circulating Water System

The circulating water system cooling towers are not part of the SBWR standard scope. A conceptual design is used for reference. The conceptual SBWR design uses a hyperbolic natural draft cooling tower. The Circulating Water System (CWS) provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the power cycle heat sink, which is the cooling tower.

The tower has a basin underneath it to collect the cooled water. The circulating water pumps are in the intake structure adjacent to the tower, and takes suction from the basin.

The CWS does not serve or support any safety function and has no safety design basis.

To prevent flooding of the turbine building, the CWS will automatically isolate in the event of gross system leakage. The circulating water pumps are tripped and the pump and condenser valves are closed in the event of a system isolation signal from the condenser area high-high level switches. A condenser area high level alarm is provided in the MCR.

A reliable logic scheme is used (e.g., 2-out-of-3 logic) to minimize potential for spurious isolation trips.

The CWS consists of the following components:

- Intake screens located in a screen house;
- Pumps;
- Condenser water boxes;
- Piping and valves;
- Tube-side of the main condenser; and
- Water-box fill and drain subsystem.

### 1.2.2.11 Station Auxiliaries

#### 1.2.2.11.1 Makeup Water System

The Makeup Water System (MWS) demineralizes water from the station water system, stores it, and transfers it to plant water systems and supply points.

The demineralization subsystem consists of cartridge filters, and a reverse osmosis and filter package. The storage and transfer system includes an outdoor makeup water storage tank, and two redundant transfer pumps. The system is housed in and controlled from the water treatment building. System components in contact with the demineralized water are stainless steel. The storage tank is freeze protected. The MWS is non-safety-related.

#### 1.2.2.11.2 Condensate Storage and Transfer System

The Condensate Storage and Transfer System (CS&TS) stores condensate grade water and transfers it to plant water systems and supply points. End users include the main condenser hotwell, CRD system, RWCU/SDC system fill, FAPCS fill, suppression and GDCS pools fill, C&FS fill, and liquid and solid radwaste system flushing.

The CS&TS includes a storage tank and transfer pumps. Components in contact with the condensate are stainless steel. The storage tank has a floating stainless steel cover and is freeze protected. A wall is built around the tank to ensure the entire tank contents is contained if there is a leak.The system is non-safety-related.

#### 1.2.2.11.3 Reactor Component Cooling Water System

The Reactor Component Cooling Water System (RCCWS) cools reactor auxiliary equipment including the Reactor Building Chilled Water System, the Drywell Cooling System, the RWCU/SDC non-regenerative heat exchangers, the FAPCS heat exchangers, and several local air coolers.

The RCCWS has two trains. Each train has two pumps, a heat exchanger, a head tank, and a chemical addition tank. The RCCWS heat exchangers are cooled by the Plant Service Water System.

Except for containment isolation, the RCCWS is non-safety-related and Seismic Category NS.

#### 1.2.2.11.4 Turbine Component Cooling Water System

The Turbine Component Cooling Water System (TCCWS) cools Turbine Building auxiliary equipment including turbine lube oil coolers, offgas condensers, generator stator and hydrogen coolers, and the instrument and service air compressors. The TCCWS is non-safety-related.

#### 1.2.2.11.5 Chilled Water System

The Chilled Water System (CWS) is made up of the Reactor Building Chilled Water System (RBCWS) and the Main Control Room Chilled Water System (MCRCWS). The RBCWS provides chilled water to the air handling units in the clean area, controlled area, and refueling area ventilation systems, the access area and change room recirculation air conditioning units, and is a backup to the RCCWS for the drywell air coolers. The MCRCWS provides chilled water to the main control room air handling units.

The RBCWS and MCRCWS each have two trains. Each train has a packaged water chiller unit with local control panel, pump, head tank, air separator, and shared chemical feed tank. The RBCWS condensers are cooled by the RCCWS and the MCRCWS condensers are air cooled by electric fans.

A chilled water systems provides chilled water for turbine and radwaste buildings. The CWS is non-safety-related and Seismic Category NS.

### 1.2.2.11.6 Oxygen Injection System

The Oxygen Injection System (OIS) adds oxygen to the condensate to suppress corrosion and corrosion product release in the C&FS. The oxygen supply consists of high pressure gas cylinders or a liquid tank. A condensate injection module is provided with pressure regulators, piping, valves and controls to depressurize the gaseous oxygen and route it to the injection modules.

### 1.2.2.11.7 Plant Service Water System

The Plant Service Water System (PSWS) cools the RCCWS and TCCWS heat exchangers. The PSWS cooling towers are not in the SBWR standard scope. A conceptual design using cooling towers for the auxiliary heat sink is used for reference purposes. The reference design for the PSWS consists of two mechanical draft cooling towers, basins, and two 100% capacity trains (50% capacity during shutdown cooling operation). Each train has two 50% capacity vertical wet pit pumps, and duplex strainers. A drain pump is also included for draining the RCCWS heat exchangers to the PSWS basin.

The towers are the multiple cell type with a two-speed reversible fan. Mechanical and electrical isolation of the cooling towers allows maintenance during full power operation. Makeup to the basins is from the station water system. The basins are normally interconnected but can be separated from each other for maintenance by using gates. Blowdown is by gravity to the natural draft cooling tower basin.

The PSWS is non-safety-related and Seismic Category NS.

### 1.2.2.11.8 Service Air System

The Service Air System (SAS) provides air for general plant use via service outlets, filter backwashing, tank sparging, and the plant breathing air system. It also serves as a backup to the Instrument Air System (IAS). The system consists of two 50% (maximum) capacity trains each with an intake air filter, compressor, aftercooler, moisture separator, and an air receiver. The breathing air subsystem includes a breathing air purifier package and an air receiver.

The system is non-safety-related and Seismic Category NS.

### 1.2.2.11.9 Instrument Air System

The Instrument Air System (IAS) supplies dry, oil-free compressed air for plant instrumentation, control systems, and pneumatic valve actuators in the various plant buildings. It consists of two 100% capacity trains each with an intake air filter, compressor, aftercooler, moisture separator, air receiver, and air dryer package.

The system is non-safety-related and Seismic Category NS.

#### 1.2.2.11.10 High Pressure Nitrogen Supply System

The High Pressure Nitrogen Supply System (HPNSS) supplies clean dry, oil-free high pressure nitrogen gas through piping from the Containment Atr spheric Control System (CACS) to meet the requirements of the main steam sys .m SRVs, ADS accumulators, and MSIVs, instruments and pneumatic valves using nitrogen in the containment. Normally the CACS supplies nitrogen gas; however, when this pressure is lost, the CACS is isolated and the HPNSS then supplies nitrogen from its bottle racks.

This system is non- safety-related and Seismic Category NS except for safety-related penetrations, and isolation valves. These components are safety-related, and Seismic Category I. The SRV ADS accumulators and piping are part of the Nuclear Boiler System.

#### 1.2.2.11.11 Auxiliary Boiler System

The Auxiliary Boiler System (ABS) supplies steam for heating of the Hot Water System (HWS) when extraction steam is not available, turbine gland sealing during startup and as a backup during normal operation, warming of the offgas preheater, and evaporation of liquid nitrogen for containment inerting.

The system consists of a package boiler and steam distribution piping and valves. It is non-safety-related.

#### 1.2.2.11.12 Hot Water System

The Hot Water System (HWS) supplies hot water for building heating. The system has two heat exchangers, two circulating pumps, and a head/surge tank. The auxiliary boiler is used to heat the water. The system supplies ventilating systems in the reactor building, turbine building, and radwaste building. It is non-safety-related.

#### 1.2.2.11.13 Hydrogen Water Chemistry System

The Hydrogen Water Chemistry (HWC) System is used, along with other measures, to reduce the likelihood of corrosion failures which would adversely affect plant availability. The function of the HWC System is to reduce the dissolved oxygen in the reactor water by the addition of hydrogen to the feedwater. This reduction has been demonstrated to be highly effective in the mitigation of the potential for intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steels.

The concentration of hydrogen and oxygen in the main steam line and eventually in the main condenser is altered during HWC system operation. This leaves an excess of hydrogen in the main condenser that would not have equivalent oxygen to combine with in the OGS. To maintain the process offgas nearer its normal constituent balance, the HWC injects a flow of oxygen upstream of the recombiner.

The HWC system is composed of hydrogen and oxygen supply systems, systems to inject hydrogen into the C&FS and oxygen into the OGS and subsystems to monitor the effectiveness of system operation.

The HWC System is non-safety-related. It is required to be safe and reliable, consistent with the requirement of using hydrogen gas.

### 1.2.2.11.14 Post-Accident Sampling System

The post accident sampling station (PASS) consists of sample holding rack, sampling rack, sample conditioning rack, local control panel, and shielding casks. All valves for PASS operaion are operated remotely. The sampling system isolation valves are operated from the main control room and all other valves are operated from the local control panel. After the sample vessel has been isolated and removed, the piping is flushed with demineralized water.

The sample holding rack has an enclosure around the sample vessel to contain any leaks of liquids or gases. The liquids drain to the radwaste system and the gases go to the reactor building exhaust system.

The PASS isolation valves are connected to a reliable source of power that will be available starting at least one hour after a LOCA or ATWS event. The isolation valves have Class 1E power and the panels and other equipment are powered with two offsite power supplies.

Gas samples are obtained from a sample line connected to the containment atmosphere monitoring system. A vacuum pump is provided to transfer the gas sample from a sample holding rack to a sampling rack.

The upper limit for activity levels in liquid and gas samples are:

- liquid samples $3.7 \times 10^4$ MBq/g (1 CI/g)
- gas samples 0.37 MBq/g (10 $\mu$Ci/g)

Means to reduce radiation exposure are provided such as, shielding, remotely operated valves, and sample transporting casks.

### 1.2.2.11.15 Process Sampling System

The Process Sampling System (PSS) collects liquid and gas samples for analysis and provides the information required to monitor plant and equipment performance and changes to operating parameters. The system samples all principal fluid process streams and consists of permanently installed sampling nozzles and sample lines, sampling panels with analyzers and associated equipment, and provisions for local grab sampling. The system is non-safety-related.

#### 1.2.2.11.16 Freeze Protection

The Freeze Protection System provides insulation, steam, and electrical heating for all external tanks and piping that may freeze during winter weather. This system is not part of the SBWR standard design and is provided here for reference purposes.

#### 1.2.2.11.17 Iron Injection System

The Iron Injection System consists of an electrolytic iron ion solution generator and equipment to inject the iron solution into the feedwater system in controlled amounts.

### 1.2.2.12 Station Electrical System

#### 1.2.2.12.1 Electrical Power Distribution System

On-site power is supplied from either the plant turbine generator, utility power grid, or an off-site power source depending on the plant operating status. During normal operation, plant loads are supplied from the main generator through the unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

The isolated phase bus connects the main generator to the generator breaker, on to the main transformer, and over to the unit auxiliary transformers. The unit auxiliary transformers power the metal clad 6.9 kV switchgear via the non-segregated phase bus. This switchgear powers some large loads and load centers consisting of 6.9 kV/480 V transformers and associated metal clad switchgear. The design includes four Class 1E 480 Vac motor control centers (MSCCs) that supply the Class 1E battery chargers and provide backup power to the Vital ac power supply.

Six individual voltage regulating transformers supply 120 Vac non-safety-related control and instrument power.

***Grounding***

The electrical grounding system is comprised of:

- an instrument grounding network for grounding of instrumentation and computer systems;
- an equipment grounding network for grounding electrical equipment (e.g., switchgear, motors, distribution panels, cables, etc.) and selected mechanical components (e.g., fuel tanks, chemical tanks, etc.);

- a lightning protection network for protection of structures, transformers and other equipment located outside buildings
- a plant grounding grid.

All grounding networks are insulated from each other and separately grounded to the plant grounding grid outside the structures. All grounding networks and equipment are low resistance grounded except the main generator, the emergency diesel generators, and the CTG, which are high resistance grounded to maximize availability. All components requiring grounding are identified and provided with grounding connections.

### 1.2.2.12.2 Direct Current Power Supply

The Class 1E dc power supply provides power to the Class 1E vital ac buses through inverters, and to 125 Vdc loads required for safe shutdown.

Each of the four divisions of Class 1E dc power is separate and independent. The dc systems operate ungrounded (with ground detection circuitry) for increased reliability. Each division has a 125 Vdc battery and a battery charger fed from its divisional 480 Vac Motor Control Center (MCC). This system is designed so that no single failure in any division of the 125 Vdc system will prevent safe shutdown of the plant.

During a total loss of off-site power, the Class 1E system is powered automatically from two non-Class 1E standby diesel generators. If these are not available, each division of Class 1E isolates itself from the non-Class 1E system, and power to safety-related loads is provided uninterrupted by the Class 1E batteries. The batteries are sized to power safety-related loads for a 72 hour period.

The Class 1E dc power supply is designed to permit periodic testing for operability and functional performance to ensure that the full operational sequence transfers power and brings the system into operation.

Non-Class 1E dc power is supplied through four non-Class 1E 480 Vac MCCs in the same manner as the Class 1E dc power (Subsection 1.2.2.3.1). Each of the two load groups receives power from two of the non-Class 1E MCCs. One MCC in each group provides power to a 125 Vdc bus through a battery charger. A 125 Vdc station battery provides backup to the supply from the battery charger.

The second MCC in each group provides power to a 250 Vdc bus through a battery charger. A 250 Vdc station battery provides backup to the supply from the battery charger.

The two non-safety-related dc busses also supply power to the non-safety-related dc-to-ac inverter discussed in Subsection 1.2.2.12.2.

### 1.2.2.12.3 Standby AC Power Supply

The non-safety-related Standby ac Power Supply consists of two diesel generators. Each diesel generator (DG) provides 6.9 KVac power to one of the two load groups whenever the main turbine generator and the normal preferred off-site power source are not operating. When operating, the standby ac power supply provides power to safety-related loads and to non-safety-related investment protection loads. Other non-safety-related loads are not powered from the standby power source.

The 6.9 kV permanent plant busses are normally energized by either the main generator or the normal preferred off-site power source. Should these power supplies fail, their supply breakers will trip and the standby power supply (diesel generators) will be automatically signalled to start. After the standby voltage and frequency reach normal values, the standby supply breakers will close. After bus voltage is reestablished, large motor loads will be sequentially started.

Each of the two DGs will start and reach full speed and voltage within two minutes after receiving a start signal. In addition, the DGs will sustain full loads within another 65 seconds. These delays are acceptable since most loads are non-safety-related. Vital safety-related loads are powered by the Standby ac Power Supply; however, these loads are powered by UPS (for ac loads) or safety-related dc power from Class 1E station batteries when normal, preferred, or standby power is not available.

### 1.2.2.12.4 Vital (Uninterruptible) Power Supply

The Class 1E vital ac power supply provides redundant, reliable power to the safety logic and control functions during normal, upset and accident conditions.

Each of the four divisions of this Class 1E vital ac power is separate and independent. Each division is powered from an inverter supplied from a Class 1E dc bus. The dc bus receives its power from a divisional battery charger and battery. Provision is made for automatic switching to an alternate Class 1E non-vital supply in case of failure of the inverter.

### 1.2.2.12.5 Instrument and Control Power Supply

The Instrument and Control Power Supply provides 120 Vac single phase power to instrument and control loads that do not require an uninterruptible power source.

### 1.2.2.12.6 Communication System

The Communications System includes a dial telephone system, a power-actuated paging facility, a sound-powered telephone system, and an in-plant radio system.

#### 1.2.2.12.7 Lighting Power Supply

The lighting systems include: the normal, standby, emergency, and security lighting systems. The normal lighting system provides illumination under all normal plant conditions, including maintenance, testing, and refueling operations. It is powered by preferred ac from the unit auxiliary non-safety-related buses. The standby lighting system supplements the normal lighting system and also supplements the emergency lighting system in selected area of the plant. The standby lighting system is normally supplied power from preferred ac power or, alternately, from the on-site standby diesel-generators. Both lighting systems are non-safety-related.

Upon loss of the normal lighting system, the emergency lighting system provides illumination throughout the plant and, particularly, areas where emergency operations are performed (e.g., main control room, battery rooms, local control stations, ingress/egress routes). It includes self-contained dc battery-operated units for exit and stair lighting. The system supplies at least 108 lux (10 foot-candles) of lighting in those areas of the plant where emergency operations could require reading printed materials or instrument scales. In other area it provides illumination levels adequate for safe ingress or egress. Inside the main control room, emergency lighting is integrated with standby lighting.

The emergency lighting is normally supplied by preferred ac powered or, alternately, the on-site standby diesel-generators. If these sources are not available, the system (excluding self-contained battery units) is supplied by Class 1E batteries through Class 1E inverters. Excluding the self-contained battery lighting units, the emergency lighting system is safety-related.

The security lighting system provides lighting for the security center, selected security areas, and the outdoor plant perimeter. The system is normally supplied power by preferred ac or, alternately, by the on-site standby diesel-generators. The security lighting system is further backed up by a dedicated security standby diesel-generator and a dedicated uninterruptible power supply. The security lighting system is non-safety-related.

### 1.2.2.13 Power Transmission

This is not part of the reference SBWR scope. Interface requirements are established for off-site power transmission.

### 1.2.2.14 Containment and Environmental Control Systems

#### 1.2.2.14.1 Containment System

The SBWR containment, centrally located in the reactor building, features the same basic pressure suppression design concept previously applied in over three decades of

BWR power generating reactor plants.The containment consists of a steel lined reinforced concrete containment structure fulfilling its design basis as a fission product barrier even at the increased pressure associated with a postulated pipe rupture.

Main features include the upper and lower drywell surrounding the RPV and a suppression chamber containing the suppression pool that serves as a heat sink during abnormal operations and accidents.

The containment is constructed as a stepped right cylinder set on the reactor building's reinforced concrete base mat. The drywell design conditions are 379 kPa gauge (55 psig) and 171°C (340°F). The suppression chamber design conditions are 379 kPa gauge (55 psig) and 121°C (250°F).

The drywell is comprised of two volumes: an upper drywell volume surrounding the upper portion of the RPV and housing the steam and feedwater piping, the SRVs, GDCS pools, main steam drain piping and upper drywell coolers; and a lower drywell volume surrounding the lower portion of the RPV, housing the FMCRDs, neutron monitoring system, equipment platform, lower drywell coolers and two drywell sumps. The drywell top opening is enclosed with a steel head removable for refueling operations.

The gas space above the suppression pool serves as the LOCA blowdown reservoir for the upper and lower drywell nitrogen and noncondensables which pass through the eight drywell-to-supression chamber vertical vents, each with 3 horizontal vents located below the suppression pool surface. The suppression pool water serves as the heat sink to condense steam released into the drywell during a LOCA or steam from SRV actuations.

Access into the upper and lower drywells is provided through a double sealed personnel lock and also an equipment hatch. The equipment hatch is removable only during refueling or maintenance outages. Access into the suppression chamber is provided by a hatch located in the safety envelope.

Prior to reactor operation, the containment atmospheric control system in conjunction with the containment purge system and the drywell cooling fans are utilized to establish an inert gas environment in the containment with nitrogen to limit the oxygen concentration. This precludes combustion of any hydrogen which might be released subsequent to a LOCA. After the containment is inerted and sealed for plant power operation, small flows of nitrogen gas are added to the drywell and the suppression chamber as necessary to keep oxygen concentrations below 4% and to maintain a positive pressure for preventing air inleakage. High pressure nitrogen is also used for pneumatic controls inside the containment to preclude adding air to the inert atmosphere.

The containment structure has the capability to maintain its functional integrity during and following the peak transient pressures and temperatures caused by the worst LOCA pipe break postulated to occur simultaneously with loss of off-site power. The containment structure is designed to accommodate the full range of loading conditions associated with normal and abnormal operations including LOCA related design loads in and above the suppression pool (including negative differential pressure between the drywell, wetwell and reactor building), and safe shutdown earthquake (SSE) loads.

The containment structure is protected from or designed to withstand fluid jet forces associated with outflow from the postulated rupture of any pipe within the containment.

The containment design considers and utilizes leak-before-break (LBB) applicability only in regard to protection against dynamic effects associated with a postulation of rupture in high energy piping. Subsection 3.6.3 and Appendix 3C describe the implementation of the LBB approach for excluding design against the dynamic effects from postulation of breaks in high energy piping. Protection against the dynamic effects from the piping systems not qualified by the exclusion from the dynamic effects caused by their failure is provided for the drywell structure. The drywell structure is provided protection against the dynamic effects of plant-generated missles (see Section 3.5).

The containment structure has design features to accommodate flooding to sufficient depth above active fuel to permit safe removal of fuel assemblies from the reactor core after a postulated design basis accident.

The containment structure is configured to channel flow from postulated pipe ruptures in the drywell to the suppression pool through vents submerged in the suppression pool which are des[illegible]d to accommodate the energy of the blowdown fluid.

The containm[illegible]ture and penetration isolation system with concurrent operation of other accide[illegible]on systems, are designed to limit fission product leakage during and follow[illegible]ostulated design basis accident (DBA) to values well below leakage calculated for allowable off-site doses.

In accordance with Appendix J to 10CFR50, periodic leak rate tests conducted at a reduced pressure below the peak calculated DBA LOCA pressure are performed to confirm containment leakage is below the design limit of 0.5% by weight per day of the containment free air volume. Special testing capabilities are provided during outages to measure local leakage, such as individual air locks, hatches, drywell head, piping, electrical and instrument penetrations. Other features are provided to measure isolation valve leakage and to measure the integrated containment leak rate. Results from the individual and integrated preoperational leak rates are recorded for comparison with subsequent periodic leak rate test results.

The design value for a maximum steam bypass leakage between the drywell and the suppression chamber through the diaphragm floor including any leakage through the suppression chamber-to-drywell vacuum breakers is limited. Satisfying this limit is confirmed by initial preoperational tests as well as by periodic tests conducted during refueling outages. These tests are conducted at differential pressure conditions between the drywell and suppression chamber that do not clear the drywell-to-suppression chamber horizontal vents.

Equipment is provided to obtain a water tight barrier between the open reactor and the drywell during refueling. This enables the reactor well to be flooded prior to removal of the reactor steam separator, dryer assembly and to facilitate underwater fuel handling operations. Piping, cooling air ducts and return air vent openings in the reactor well platform must be removed, vents closed and sealed watertight before filling the reactor well with water. The refueling bellows assembly is provided to accommodate the movement of the vessel caused by operating temperature variations and seismic activity.

Containment isolation is accomplished with inboard and outboard isolation valves on each piping penetration which are signaled to close on predefined plant parameters. Systems performing a post LOCA function are capable of having their isolation valves reopened as needed.

Drywell coolers are provided to remove heat released into the drywell atmosphere during normal reactor operations.

The Flammability Control System provides ignitors located throughout both the drywell and suppression chamber to prevent any high-energy-release recombinant reactions potentially developing within the containment following a LOCA.

#### 1.2.2.14.2 Containment Vessel

The containment vessel is a reinforced stepped cylindrical concrete vessel (RCCV). The RCCV supports the upper pools whose walls are integrated into the top slab of the containment to provide structural capability for LOCA and testing pressures.

#### 1.2.2.14.3 Containment Internal Structures

The containment system's principal internal structure consists of the structural barrier separating the drywell from the suppression chamber. This barrier is comprised of the suppression chamber ceiling (diaphragm floor) and the inboard wall (vertical vent wall) separating the drywell from the suppression chamber. Both of these structural components are designed as steel structures filled with insulating concrete to minimize long-term heat transfer from drywell to wetwell. The vertical vent wall also provides a durable attachment point for the RPV horizontal stabilizers.

An all-steel reactor shield wall of appropriate thickness is provided, which surrounds the RPV to reduce gamma shine on drywell equipment during reactor operation and protect personnel during shutdowns for maintenance and inservice inspections. The RPV insulation is supported from the internal surface of the reactor shield wall. The reactor shield wall is supported on top of the pedestal support structure.

Various drywell piping and equipment support structures are provided to support electric and instrument cable trays, drywell coolers, air distribution ductwork, steam and feedwater piping, and SRV discharge piping. Support is provided for isolation valves and piping of the ICS and PCCS. This steel structure also supports access stairs, walkways, railings and gratings. Monorails are suspended from the ceiling of the drywell for hoists to work on NSSS equipment.

### 1.2.2.14.4 Passive Containment Cooling System

The PCCS maintains the containment within its pressure limits for design basis accidents such as a LOCA. The system is passive with no components that move.

The PCCS consists of three low pressure, totally independent loops, each containing a steam condenser (passive containment cooling condenser) that condenses steam on tube side and transfers heat to water in a large cooling pool (IC/PCCS pool), which is vented to atmosphere.

Each PCCS condenser is located in a subcompartment of the IC/PCCS pool, and all pool subcompartments communicate at their lower ends to enable full use of the collective water inventory, independent of the operational status of any given PCCS loop.

Each loop which is open to the containment, contains a drain line to the GDCS pool, and a vent discharge line the end of which is submerged in the pressure suppression pool.

The PCCS loops are driven by the pressure difference created between the containment drywell and the suppression pool during a LOCA so require no sensing, control, logic or power actuated devices for operation.

The PCCS is classified as safety-related and Seismic Category I.

Each of the three PCC condensers is designed for 10 MWt capacity. Together with the pressure suppression containment system, the three PCC condensers limit containment pressure to less than its design pressure for at least 72 hours after a LOCA without make-up to the IC/PCC pool.

The PCC condensers are in a closed loop extensions of the containment pressure boundary. Therefore, there are no containment isolation valves and they are always in "ready standby".

The PCCS can be periodically pressure-tested as part of overall containment pressure testing. Also, the PCC loops can be isolated for individual pressure testing during maintenance.

During refueling outages, the in-service inspection (ISI) of PCC condenser can be performed, if necessary, because ultrasonic testing of tube-to-heater welds and eddy current testing of tubes can be done with PCCs in place. The PCC condenser is located in the IC/PCC pool.

### 1.2.2.14.5 Containment Atmospheric Control System

The Containment Atmospheric Control System (CACS) is designed to establish and maintain an inert atmosphere within the containment during all plant operating modes except during plant shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The objective of the system is to establish conditions that help preclude combustion of hydrogen and thereby prevent damage to safety-related equipment and structures.

The CACS does not perform any safety-related functions except for its containment isolation function. Failure of the CACS does not compromise any safety-related system or component nor does it prevent a safe shutdown of the plant. (The inerted conditions that CACS establishes are safety-related, however.)

CACS establishes an inert atmosphere (i.e., an oxygen concentration ≤ 4% by volume) throughout the containment following an outage (or other occasions when the containment has become filled with air) and maintains it inert during normal conditions. The system maintains a slight positive pressure in the containment to prevent air (oxygen) in-leakage.

CACS is comprised of a pressurized liquid nitrogen storage tank, a steam heated main vaporizer for large nitrogen flow, electric heater for vaporizing makeup flow, two injection lines, an exhaust line, a bleed line, associated valves, controls, and instrumentation. All CACS components are located inside the reactor building except the liquid nitrogen storage tank and the steam-heated main vaporizer which are located in the yard.

The first of the injection lines is used only for makeup; it includes an electric heater to vaporize the nitrogen and to regulate the nitrogen temperature to acceptable injection temperatures. Remotely operated valves together with a pressure-reduction valve enable the operator to accomplish low rates of nitrogen injection into the drywell and suppression chamber airspace.

The second injection line is used when larger inerting flow rates are required. This line takes vaporized nitrogen from the steam-heated main vaporizer, uses remotely operated valves together with a pressure-reduction valve and injects nitrogen at points in common with makeup supply. The inerting and makeup lines converge to common injection points in the lower drywell and suppression chamber airspace.

The CACS includes an exhaust line leading from the upper drywell at the opposite side from the injection points. The discharge line connects to the reactor building HVAC system exhaust where exit gases are served by exhaust fans, filters, and radiation monitors before being diverted to the plant stack. A small bleed line bypassing a short portion of the main exhaust line, upstream of the fans, filters, and stack monitors, is also provided for manual pressure control of the containment during normal reactor heatup.

Redundant containment isolation valves provided in the inerting, makeup, exhaust and bleed lines close automatically upon receipt of an isolation signal from the LD&IS.

Upstream of the pressure-reduction valve in the makeup line, a small branch line is provided and connected to the HPNSS. This line is used for the initial charging of the HPNSS and makeup to keep the HPNSS changed with nitrogen during normal plant operation.

During plant startup, a large flow of nitrogen from the liquid nitrogen storage tank is vaporized by the steam-heated vaporizer and injected into the drywell and the suppression chamber. It is then mixed into the containment atmosphere by the drywell cooling fans. The exhaust line is kept open to displace containment resident atmosphere with nitrogen. Once the desired concentration of nitrogen is reached, the exhaust line is allowed to close. When the required inerted containment operating pressure is attained, the inerting process is terminated by the closure of the nitrogen supply shutoff valve and inerting isolation valves. The system is designed to inert the containment to ≤ 4% oxygen by volume within four hours.

Following shutdown, the containment atmosphere is de-inerted to allow safe personnel access inside the containment. Breathable air from the Reactor Building HVAC System is injected to the drywell and suppression chamber air space through the inerting injection line. The incoming air displaces containment gases (mostly nitrogen) into the exhaust line. Vented gases are served by the Reactor Building HVAC system exhaust fans, filters, and radiation detectors before being diverted to the plant stack. The system is designed to de-inert the containment to an oxygen concentration of ≥18% within four hours.

#### 1.2.2.14.6 Drywell Cooling System

The Drywell Cooling System (DCS) consists of four fan coil units (FCUs), two located in the upper drywell, and two in the lower drywell. The system uses the FCUs to deliver cooled air/nitrogen to various areas of the upper and lower drywell through ducts/diffusers. The DCS is a closed loop recirculating air/nitrogen cooling system where no outside air is introduced into the system except when the containment is open. The DCS system is manually controlled from the MCR. During normal plant operation, the DCS is cooled by the RCCWS. During shutdown operation, the DCS is cooled by the CWS to facilitate obtaining cooler temperatures. The CWS water is also used as a backup to RCCWS water when the latter is not available.

Through the entire plant operating range, from startup to full load condition or from full load to shutdown, the DCS performs the following functions:

- Maintains temperature and humidity in the upper and lower drywell spaces within specified limits during normal operation.
- Maintains the RPV support skirt temperature within specified limits to satisfy structural requirements.
- Accelerates drywell cooldown during the period from hot reactor shutdown to cold shutdown.
- Aids in complete purging of nitrogen from the drywell during shutdown.
- Maintains a habitable environment for plant personnel during plant shutdowns for refueling and maintenance.
- Limits drywell temperature during loss of preferred power (LOPP).

The DCS is designed to maintain the following conditions in the upper and lower drywell during normal and plant shutdown modes of operation:

Normal plant operation:

- Average dry bulb temperature: 57°C (135°F)
- Maximum temperature of ambient atmosphere in each drywell zone: 66°C (150°F)

Plant shutdown:

- Average dry bulb temperature: 26°C (77°F)

There are two direct-drive fans in each FCU. Each FCU motor is controlled manually from the MCR. Indicator lights show the status of each unit. Failure of an FCU with consequent temperature rise in the discharge stream or loss of flow actuates an alarm in the MCR.

Each upper drywell FCU has a cooling capacity of 50% of the upper drywell design cooling load under normal plant operating conditions. Likewise, each lower drywell FCU has a cooling capacity of 50% of the lower drywell design cooling load. All FCUs normally operate. Each FCU is composed of a cooling coil and two fans downstream of the coil. One FCU is supplied by RCCWS loop A and the other by RCCWS loop B. One of the fans operates while the other is on standby status and will automatically start upon loss of the lead fan. During normal operation, if both fans of an FCU are out of commission, or the unit is not in service for some other reason, then both fans on the other unit in the area (upper or lower drywell) operate and the cooling supply transfers to the CWS.

Cooled air/nitrogen leaving the FCUs enter a common plenum and is distributed to the various zones in the drywell through distribution ducts. Return ducts are not provided; the FCUs draw air/nitrogen directly from the upper or lower drywell.

A condensate collection pan is provided with each FCU. The condensate collected from all FCUs in the upper and the lower drywell is piped to an LD&IS flow meter to measure the condensation rate of unidentified leakages.

#### 1.2.2.14.7 Flammability Control System

The Flammability Control System (FCS) is designed to limit the concentration of oxygen in a potentially hydrogen-rich post-accident containment atmosphere by controllably burning hydrogen at low levels of oxygen inside the containment.

The FCS consists of divisionally assigned low power consumption igniter assemblies strategically intermixed throughout the containment including the upper and lower drywell cavities, and wetwell air space, and powered by Class 1E divisional power.

The FCS is controlled from the MCR. Prior to the postulated design basis LOCA, the containment is maintained inert at ≤ 4% oxygen volumetric concentration by the CACS. The FCS automatically initiates 24 hours after receipt of a LOCA signal for the controlled ignition of hydrogen with oxygen. Once initiated, igniters will continue to operate unless manually stopped by the operator. Manual FCS initiation is also possible from the MCR.

During normal plant operation, the CACS provides containment atmosphere oxygen level monitoring. During FCS operation, post-accident oxygen level monitoring is provided by the Containment Atmospheric Monitoring System (CAMS).

The FCS is designed and qualified as a safety-related and Seismic Category I system. All required FCS components are designed and qualified to withstand the adverse environmental conditions resulting from DBA LOCA.

#### 1.2.2.15 Structures and Servicing Systems

##### 1.2.2.15.1 Cranes, Hoists, and Elevators

Large bridge cranes are provided in the turbine building and for the refueling floor. A bridge crane is also installed in the radwaste building. Miscellaneous hoists and monorails are installed in the reactor, turbine and other buildings as necessary for maintenance and replacement of equipment. Elevators are installed in the reactor and turbine buildings.

##### 1.2.2.15.2 Heating Ventilating and Air Conditioning

***Reactor Building HVAC Systems***
These systems are the Clean Area Ventilation System (CLAVS), the Controlled Area Ventilation System (CONAVS), Control Room Envelope HVAC (CREHVAC), and the Refueling and Pool Area Ventilation System (REPAVS). A common intake is used for these systems. With the exception of containment isolation components, the systems are non-safety-related and Seismic Category NS.

The CLAVS includes redundant supply fans, redundant air conditioning units (with air mixing plenum, filters, heating and cooling coils, and humidifier), dampers, and ducting. The system also includes redundant return/exhaust, battery room exhaust, and smoke removal fans. Local cooling/heating coils and fans are provided for the main entrance area and the access and change room area.

The CONAVS has two main trains each including a supply fan, air conditioning units (with filters and heating and cooling coils), and an exhaust fan. Two redundant exhaust fans are provided for the safety envelope area, and local recirculating systems (including redundant fans, and cooling/heating coils) are provided for the FAPCS, RWCU/SDC, RCCW, main steam tunnel, and CRD pump rooms. CONAVS also includes a separate containment purge and exhaust subsystem with purge supply and exhaust filters, redundant supply and exhaust fans, and main stack radiation monitors.

All CLAVS and CONAVS equipment is non-safety-related with the exception of the isolation dampers and ducting that penetrate the safety envelope.

CREHVAC serves the MCR, technical support center (TSC), computer room, and adjacent rooms and includes redundant supply fans, air conditioning units (with air mixing plenum, filters, heating and cooling coils, and humidifier), and exhaust fans. Two utility exhaust fans are also provided as well as a supplementary filtration unit with HEPA and charcoal filters and redundant exhaust fans for removal of airborne hazardous materials. With the exception of isolation dampers and ducting for the MCR/TSC computer areas, the system is non-safety-related.

Refueling and Pool Area Ventilation System (REPAVS) has two full capacity supply fans, air conditioning units with filters, cooling and heating coils, and exhaust fans. It is non-safety-related except for dampers and ducting associated with refueling floor isolation.

***Turbine Building HVAC***

The turbine building ventilation system includes an intake plenum and dampers and two 100% capacity supply trains with an air conditioning unit (filters, heating and cooling coils, and humidifier). The turbine building chilled water system provides chilled water to local unit coolers and outside air intake coils when required. Two redundant exhaust fans are provided. Local unit coolers and fans are provided in areas with high local heat loads. The system is non-safety-related.

***Other Building HVAC***

Ventilation for other buildings includes the radwaste building, electrical building, service building, water treatment building, administration building, guard house, etc. All these systems are non-safety-related, of conventional design and typically include redundant supply and exhaust fans, and air conditioning units. The radwaste building and hot machine shop ventilation systems also include additional filtration and airborne radioactivity monitoring equipment.

### 1.2.2.15.3 Fire Protection System

The Fire Protection System (FPS) includes the fire protection water supply system, yard piping, water sprinkler, standpipe and hose systems, a foam system, smoke detection and alarms systems, and fire barriers.

The water supply system includes a motor-driven pump and a backup diesel-engine driven pump. Yard piping supplies fire water to all buildings. Fire hydrants are located throughout the site. Standpipes are provided within buildings as well as automatic sprinkler and deluge systems. Foam fire suppression systems are provided for the standby diesel generator and day tank rooms, outdoor diesel fuel oil storage tanks, and the turbine lube oil system and storage tanks. Smoke and heat detectors are located throughout the various buildings and are controlled by local panels and provide remote indication in the MCR. Fire barriers (typically three-hour rated), including penetration seals, doors, and fire dampers are provided wherever separation of redundant safety-related equipment is required.

The FPS is non-safety-related. The diesel-driven fire pump, its suction line, a portion of the yard piping and connecting piping serving safety-related areas are designed to remain functional after an SSE.

### 1.2.2.15.4 Equipment and Floor Drainage System

The Equipment and Floor Drainage System (EFDS) serves the plant building with floor and equipment drains and consists of the following drain subsystems: clean, low conductivity waste (LCW), high conductivity waste (HCW), detergent, and chemical waste. All potentially radioactive drains are routed to the Radwaste Management System for processing.

Each subsystem includes sumps, sump pumps, piping and valves, and level instrumentation and controls.

The EFDS is non-safety-related except for containment penetrations and isolation valves.

### 1.2.2.15.5 Reactor Building

The reactor building houses the reactor system, reactor support and safety systems, containment, refueling and spent fuel storage areas and equipment, main steam tunnel, MCR and other control areas, auxiliary area, liquid waste processing area, health physics, laboratories, security and access control areas.

The reactor building structure is integrated with that of a stepped cylindrical reinforced concrete containment vessel (RCCV); the RCCV is located on a common basemat and surrounded by three concentric boxes: the inner box (safety envelope), the intermediate steel frame, and the outer box. The inner and outer boxes are made of reinforced concrete shear walls and the intermediate steel frame is made of structural steel framework with non-structural walls as required for radiation shielding, separation, etc. The building is partially embedded.

All SBWR safety-related equipment is housed in the reactor building safety envelope, main steam tunnel, and pools located beneath the operating floor, with the non-safety-related systems and areas (including the MCR) surrounding this envelope. The safety envelope is leaktight for holdup and decay of fission products that may leak from the containment after an accident. This holdup capability decreases releases to the atmosphere. The building and systems are also arranged to separate clean and potentially contaminated areas, with separate stairway and elevator service for each area.

On the upper levels of the reactor building is the refueling area which contains the spent and new fuel pools, cask loading area, isolation condenser/passive containment cooling system pools, other pools and storage areas, and refueling and fuel handling systems. A bridge crane is installed that operates along the length of the floor and services a large equipment hatch that is provided at grade with a shaft allowing communication with all elevations up to the refueling floor.

A plant stack is located on the reactor building and rises above the top of the building. The stack is of steel shell construction supported by an external steel tubular framework. The stack vents the reactor building. The reactor building is a safety-related and Seismic Category I structure.

#### 1.2.2.15.6 Turbine Building

The turbine building encloses the turbine-generator, main condenser, condensate and feedwater systems, condensate purification system, turbine-generator support systems, and bridge crane.

The turbine building is a reinforced concrete structure up to the turbine operating deck, and steel frame and metal siding thereafter. It is built at grade. Shielding is provided for the turbine on the operating deck. The turbine-generator and condenser are supported on spring type foundations. The turbine building is a non-safety-related structure.

#### 1.2.2.15.7 Radwaste Building

The radwaste building houses tanks and processing equipment, storage areas, a laundry room, a control room and health physics area, a truck bay, and other support facilities. A pipe tunnel connects the radwaste building to the turbine and reactor buildings. Space is included for storage of dry active waste. The structure up to grade is reinforced concrete (first story), and has a structural steel framework with metal siding and a metal roof above that. The reinforced concrete portion of radwaste building below grade is designed to the requirements of Regulatory Guide 1.143, and the balance of the structure is Seismic Category NS.

#### 1.2.2.15.8 Other Building Structures

Other facilities include the electrical building, the service building, the service water and fire building, mechanical draft cooling towers, the water treatment building, gate houses, guard house, an administration building, a training center, sewage treatment plant, warehouse, and hot and cold machine shops. These are all of conventional size and design.

The electrical building houses the two non-safety-related standby DGs. It is a reinforced concrete structure. It is non-safety-related and Seismic Category NS.

The service water and fire building houses the PSW pumps and fire pumps, and associated water storage, piping and valves. It is a concrete foundation steel frame building with metal siding and metal roof. It is non-safety-related and Seismic Category NS.

The water treatment building is a conventionally sized and designed building.

#### 1.2.2.16 Intake Structure and Servicing Equipment

##### 1.2.2.16.1 Intake and Discharge Structure

The intake and discharge structure (which is the reference design only) is adjacent to the natural draft cooling tower and houses the circulating water pumps, isolation valves, water treatment equipment, and associated electrical power and controls equipment. The structure is of conventional reinforced concrete construction. A traveling screen and trash rake system is installed to prevent debris from entering into the circulating water system. The structure and systems are not safety-related. Blowdown from the cooling tower basin is via a blowdown line to the site water source. The intake and discharge structure are provided by the applicant.

##### 1.2.2.16.2 Cooling Tower

The conceptual SBWR cooling tower is a single hyperbolic natural draft cooling tower that cools circulating water. The reinforced concrete tower is located atop a cooling tower basin. The tower system is equipped with drift eliminators and a winter bypass line for cold weather operation. It is not safety-related.

#### 1.2.2.17 Yard Structures and Equipment

##### 1.2.2.17.1 Oil Storage and Transfer Systems

The major components of this system are the fuel-oil storage tank, pumps, and day tanks. Each standby DG has its own individual supply components. Each fuel-oil pump is controlled automatically by day-tank level and feeds its day tank from the storage tank.

##### 1.2.2.17.2 Site Security

The site security system includes fencing, E-field intrusion detection systems, closed circuit television system, site access control equipment (portal monitors, identification equipment), an electronic lock/cardreader building access control system, vehicle inspection bays, and monitoring and control computers and stations.

### 1.2.3 COL License Information

The applicant shall provide necessary design information on the cooling tower, intake structure, and discharge structure.

### 1.2.4 References

None.

# ATTACHMENT 3

# OVERVIEW OF SBWR PERFORMANCE AND METHODS

CONTAINS DRAFT PRESENTATION FOR REASSESSMENT BY BS SHIRALKAR

GE Nuclear Energy

# Overview of SBWR Performance & Methods

*B.S.Shiralkar*

*June 21, 1994*

# *Overview of SBWR Performance & Methods*

### *Major SBWR Differences*

| *Feature* | *Affects* |
|---|---|
| Natural circulation | Stability, transient performance |
| Close coupling between reactor and containment | ECCS, long term containment performance |
| Passive heat removal | ECCS, long term containment response |

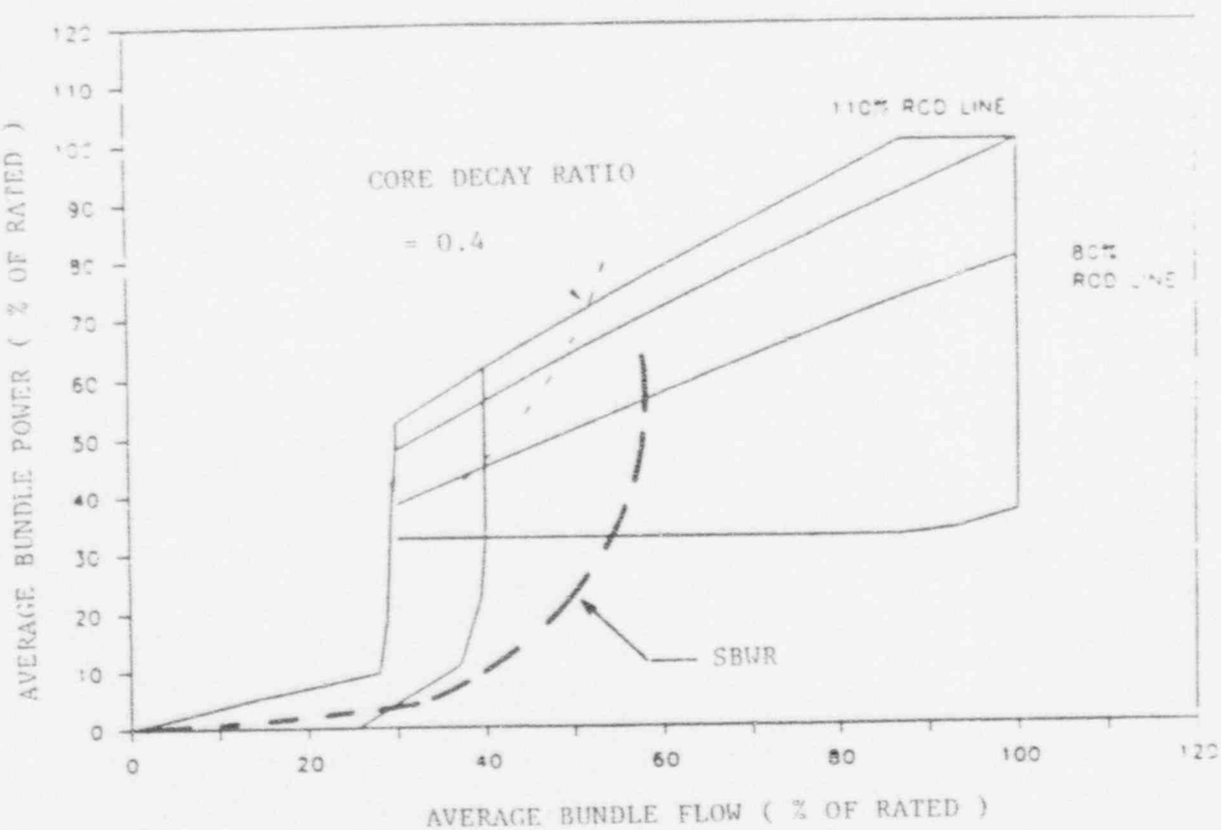
# *Overview of SBWR Performance & Methods*

***Stability Performance***

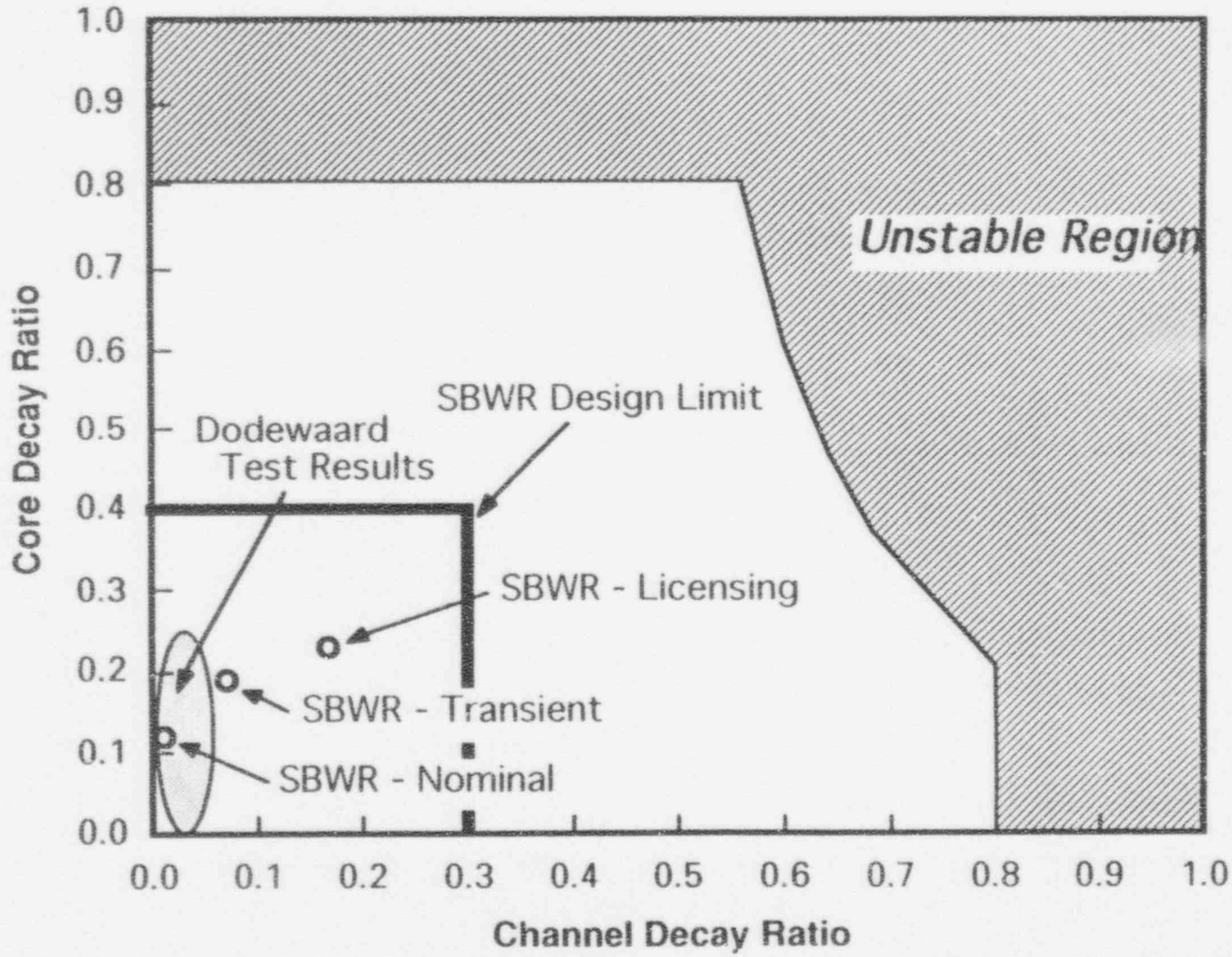
- Chimney height controls core flow rate
- SBWR designed to operate at significantly higher flow/power ratio than ABWR
- Stability assured in normal operation by designing to low core and channel decay ratios
- Scram protection prevents unstable operation during transients
- During ATWS conditions, stability is assured by :
  - Automatic ARI/FMCRD run–in
  - Automatic feedwater runback
  - Automatic boron injection

# Overview of SBWR Performance & Methods

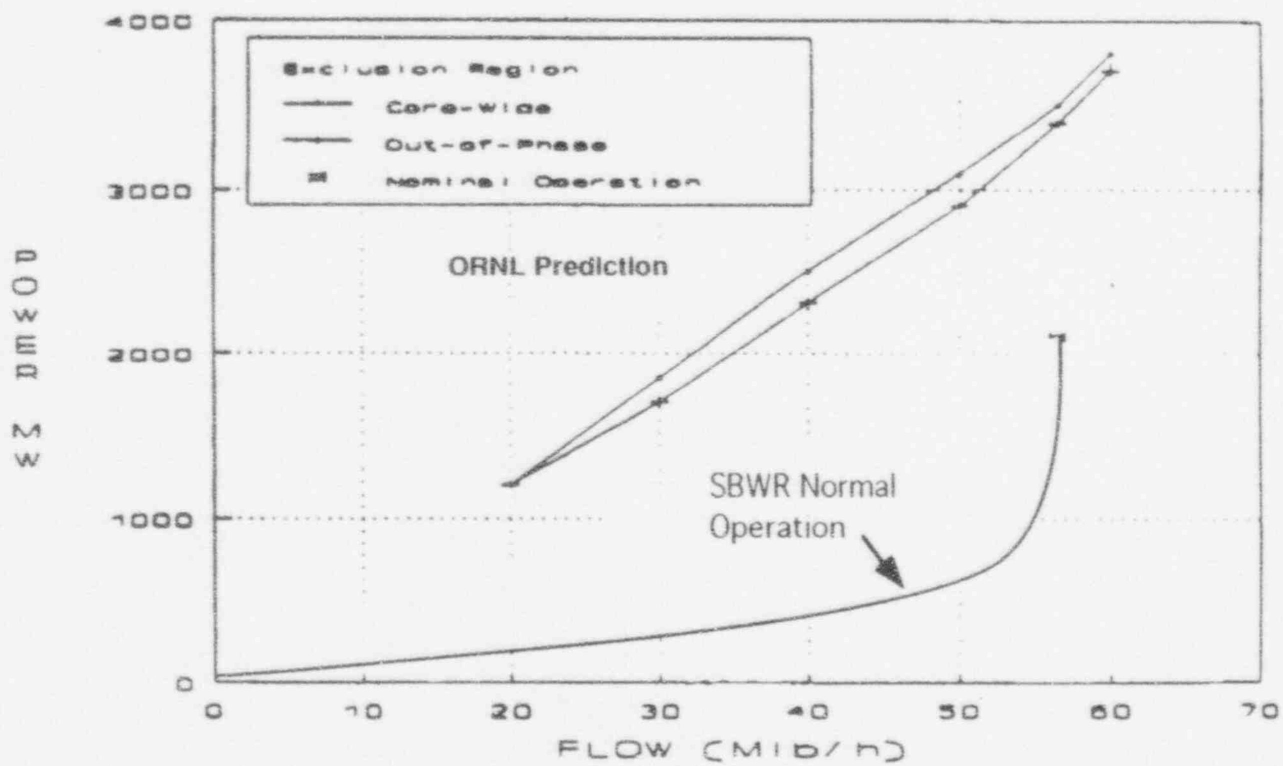
## Comparison of SBWR and BWR/5 Power/Flow Ratio



## SBWR Stability Performance



# *Stable Normal Operation*



# *Overview of SBWR Performance & Methods*

### *Transient Performance*

- BWR transients characterized by :
  - Front end coupled neutronic/thermal hydraulic transient, which leads to a new steady state or scram
  - Long term inventory maintenance
- Classes of transients analyzed :
  - Decrease in Reactor Water Temperature
  - Increase in Reactor Pressure
  - Decrease in Reactor Inventory
- Large two–phase chimney volume
  - Reduces pressurization rates
  - Increases level swings
- Load rejection without bypass is limiting ($\Delta$CPR =0.19)
- No relief valve lift for transients
- Thermal margins greater than 15%

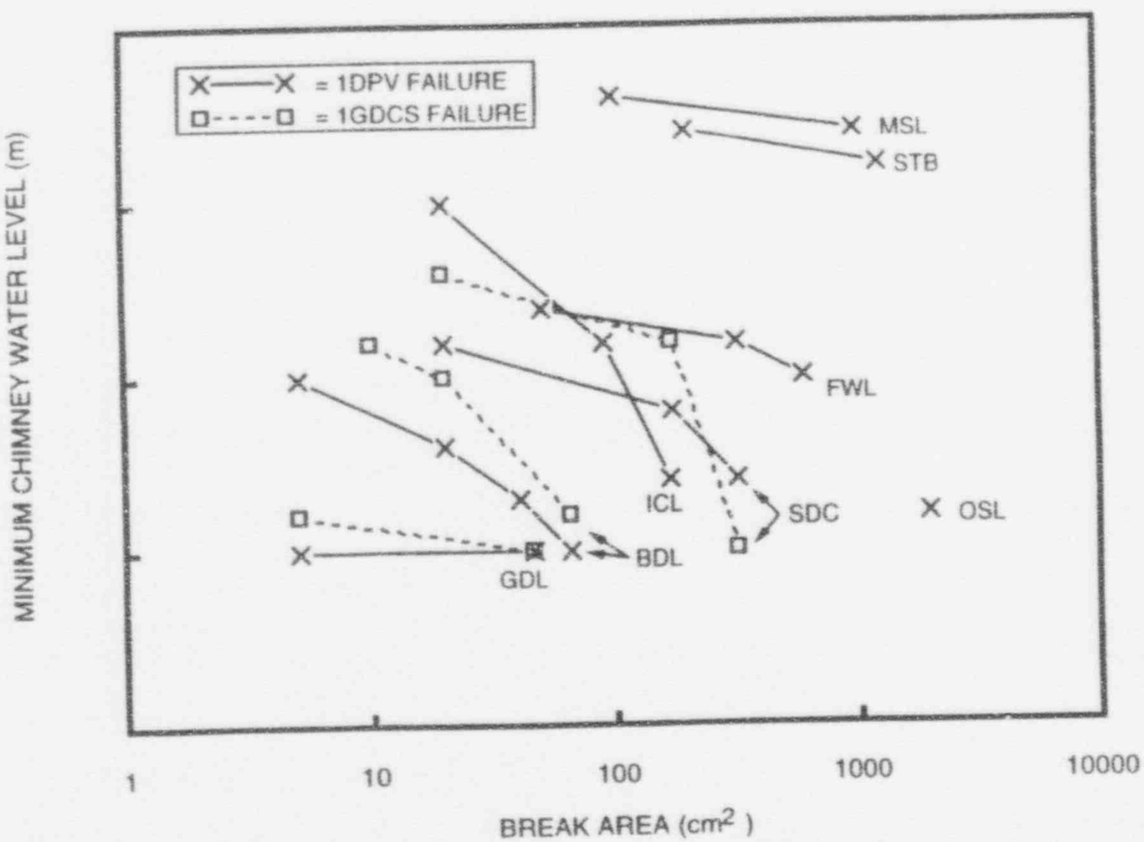
# *Overview of SBWR Performance & Methods*

### *ECCS Performance*

- Limiting breaks are GDCS line and Bottom Drain Line
- No core heatup for any break / single failure combination
- Long term makeup from GDCS flow for high breaks (steam line)
- Long term makeup from suppression pool via equalization line for GDCS line break or bottom drain line break
  - Lower drywell overflows to suppression pool

# Overview of SBWR Performance & Methods

## Relative Performance for Various Breaks

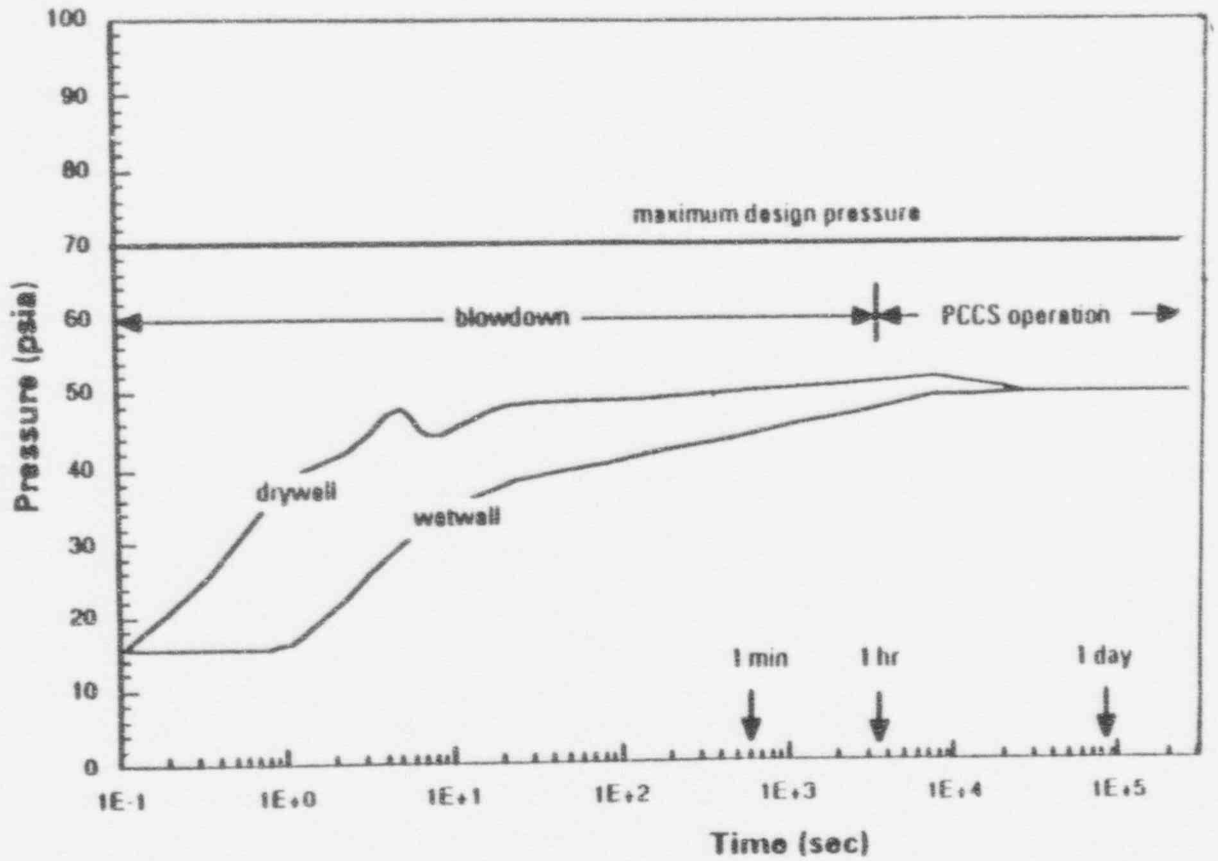


# *Overview of SBWR Performance & Methods*

## *Containment Performance*

- Short term response similar to ABWR
  - Horizontal vents / SRVs discharge to suppression pool
  - Short term pressures not limiting
- Long term decay heat removal by PCCS
  - Flows driven by gravityand small pressure differences
  - Heat removal is from drywell; drywell to wetwell leakage needs to be tightly controlled
  - Pressures/temperatures maintained within limits for 72 hours without operator actions or active systems

# Containment Response to LOCA



# Overview of SBWR Performance & Methods

### SBWR Analysis Methods

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

# *Overview of SBWR Performance & Methods*

### *TRACG Application to SBWR*

- Application of TRACG for safety analysis under review by NRC
  - All BWRs, ABWR, SBWR
  - LOCA (ECCS and containment), transients, stability, ATWS, RIA
- Three LTRs submitted 2/93 for :
  - Models (generic model description : thermal hydraulics, kinetics, numerical solution,etc.)
  - Qualification (database; generic BWR and SBWR)
  - SBWR application (analysis conditions, statistical methodology, results)
- SBWR is lead application
  - No submittals yet for generic BWR applications
- Why TRACG for SBWR ?
  - Current design codes not directly applicable
  - TRACG ideal for evolving conceptual design (not 'hard–wired' like design codes)

## *Overview of SBWR Performance & Methods*

  - NRC familiarity & acceptance
    - TRACG accepted by NRC/ACRS for benchmarking SAFER and for BWROG stability solution analysis
  - Cost effective– no need to modify and get approval for a number of design codes

***Status***

- LTR contents presented to NRC and BNL reviewers (June 93)
- Initial NRC review completed
- First set of questions received on all 3 LTRs (except containment portion)
  - Responses to questions by July

***Overall Prognosis***

- TRACG review proceeding as expected
  - Critical path is resolution of SBWR related testing needed for Qualification LTR approval

# *Overview of SBWR Performance & Methods*

*Summary*

- SBWR has large margins
  - Stability decay ratios < 0.2
  - Transient thermal margins > 15%
  - No heatup for LOCAs
  - Containment pressures below design limits by > 15%
- Margins significantly improved if active (non–safety grade) systems available
- Impact of uncertainties should be evaluated in the context of these margins
- TRACG used for most analysis
  - One major code for qualification, application and review

# ATTACHMENT 4

# GIST TEST

CONTAINS DRAFT PRESENTATION FOR REASSESSMENT BY PF BILLIG

# GRAVITY DRIVEN COOLING SYSTEM TEST PROGRAM (GIST)

**SBWR TECHNOLOGY REASSESSMENT**

**JUNE 21, 1994**

# OUTLINE

- PROGRAM DESCRIPTION
- GIST TEST FACILITY
  - DESIGN
  - GIST/SBWR DIFFERENCES
- TESTS AND RESULTS

# PROGRAM OBJECTIVES

- DEMONSTRATE THE TECHNICAL FEASIBILITY OF THE GDCS CONCEPT
- PROVIDE A DATABASE TO QUALIFY TRACG FOR ABWR ACCIDENT ANALYSES

# PROGRAM DESCRIPTION

- SPONSORED BY U.S.. DEPARTMENT OF ENERGY
- DEMONSTRATION OF GRAVITY DRIVEN ECCS CONCEPT
  - CREDIBLE DEMONSTRATION OF SBWR LOCA RESPONSE
  - LOW PRESSURE BLOWDOWN AND REFLOOD
  - RPV LEVEL VS TIME IS KEY EXPERIMENTAL OUTPUT
- 1/508 SECTOR-SCALED MOCK-UP OF 1987 SBWR CONFIGURATION
- INTEGRATED SYSTEMS TEST OF RPV AND CONTAINMENT

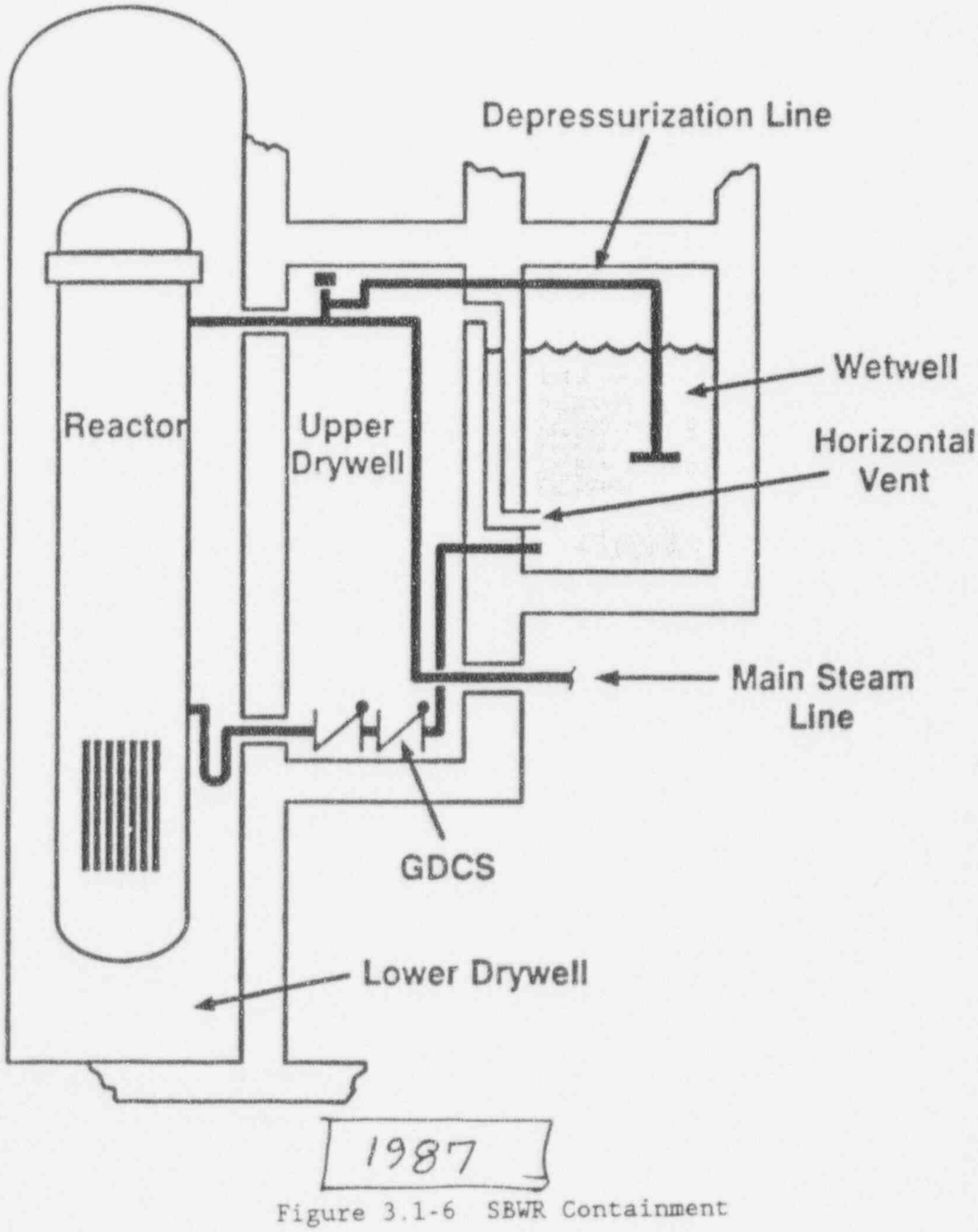
# GIST Design

**Sector Scaling of SBWR Design**

| | |
|---|---|
| • **Full Scale:** | **1/508 Scale:** |
| - Time | - Areas |
| - Pressures | - Volumes |
| - Temperatures | - Core power |
| - Differential Pressures | - Mass flowrates |
| - Elevations | |
| - Water Levels | |
| - Flow Velocities | |
| | |
| • **Total Height** | **90.1 ft** |

# GIST Design (cont.)

- **Volumes:**
  - Wetwell — $736\ ft^3$
  - Upper Drywell — 421 ft
  - Lower Drywell — 45 ft
- **Reactor Vessel**
  - Length — 61.7 ft
  - Number of heater rods — 45
  - Maximum core power — 145 KW
  - Rated pressure — 180 psig
- **No. of Data Sensors** — 120



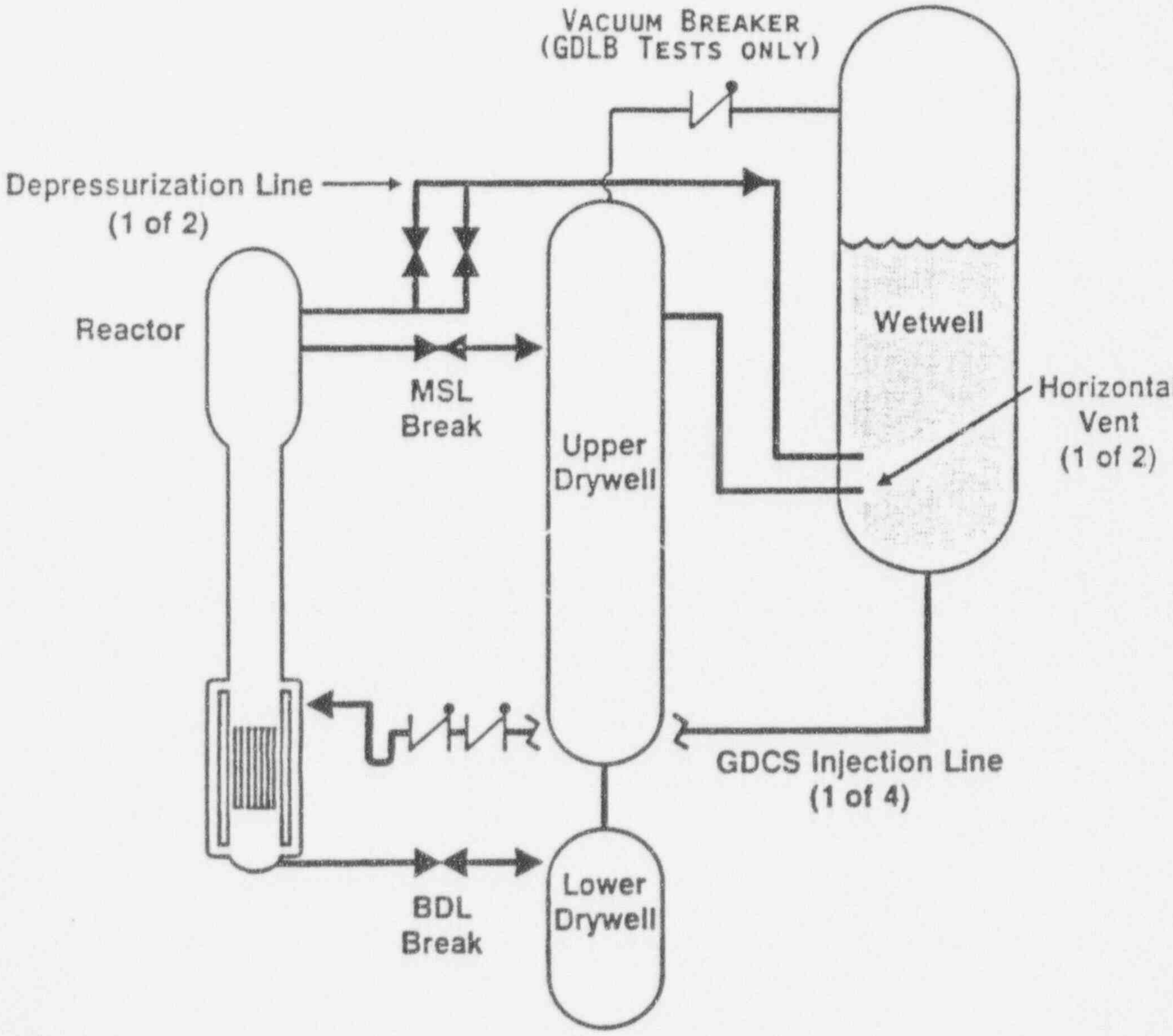

1987

Figure 3.1-6 SBWR Containment

GRAVITY-DRIVEN COOLING SYSTEM TEST PROGRAM

SCHEMATIC OF GIST

(GDCS INTEGRATED SYSTEMS TEST)

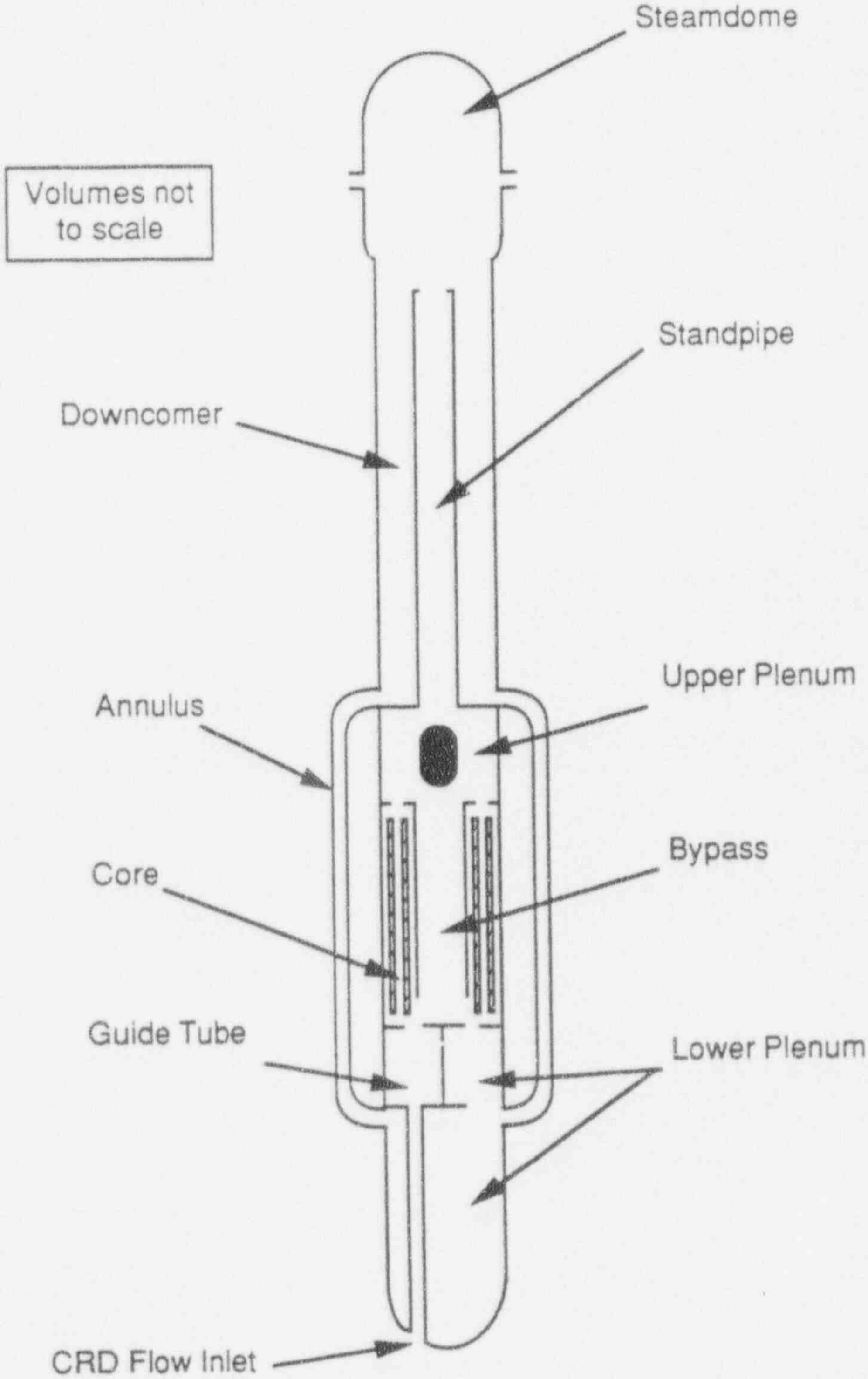



GDCS.WP

# GRAVITY-DRIVEN COOLING SYSTEM TEST PROGRAM

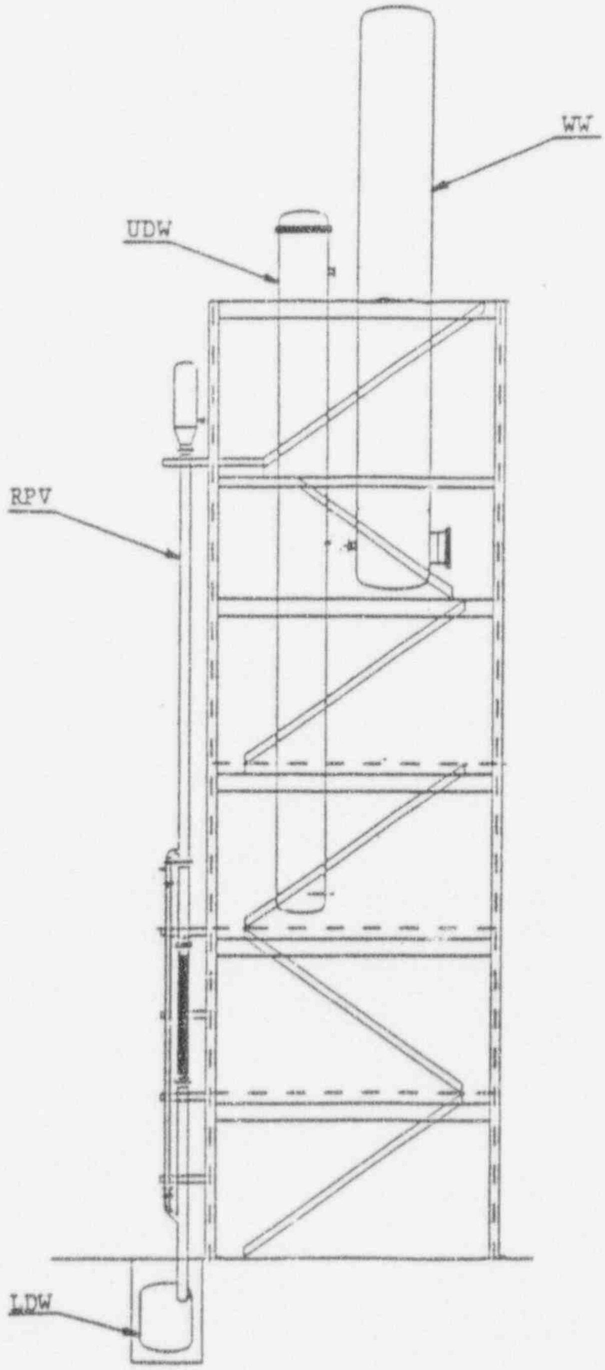
## GIST TEST FACILITY

## "REACTOR" PRESSURE VESSEL




GDCS.WP

## GIST TEST FACILITY



# SBWR - GIST DIFFERENCES

- DIFFERENCES EXIST DUE TO:
  - SBWR DESIGN CHANGES 1987-1993
  - FACILITY DESIGN REQUIREMENTS AND COMPROMISES
- DIFFERENCES MUST BE IDENTIFIED AND RECONCILED TO ASSURE THAT:
  - IMPORTANT PHENOMENA ARE NOT EFFECTED
  - NON-REPRESENTATIVE SYSTEM INTERACTIONS ARE NOT PRESENT
  - REPRESENTATIVE SYSTEM INTERACTIONS ARE PRESENT

# SBWR - GIST Difference Identification

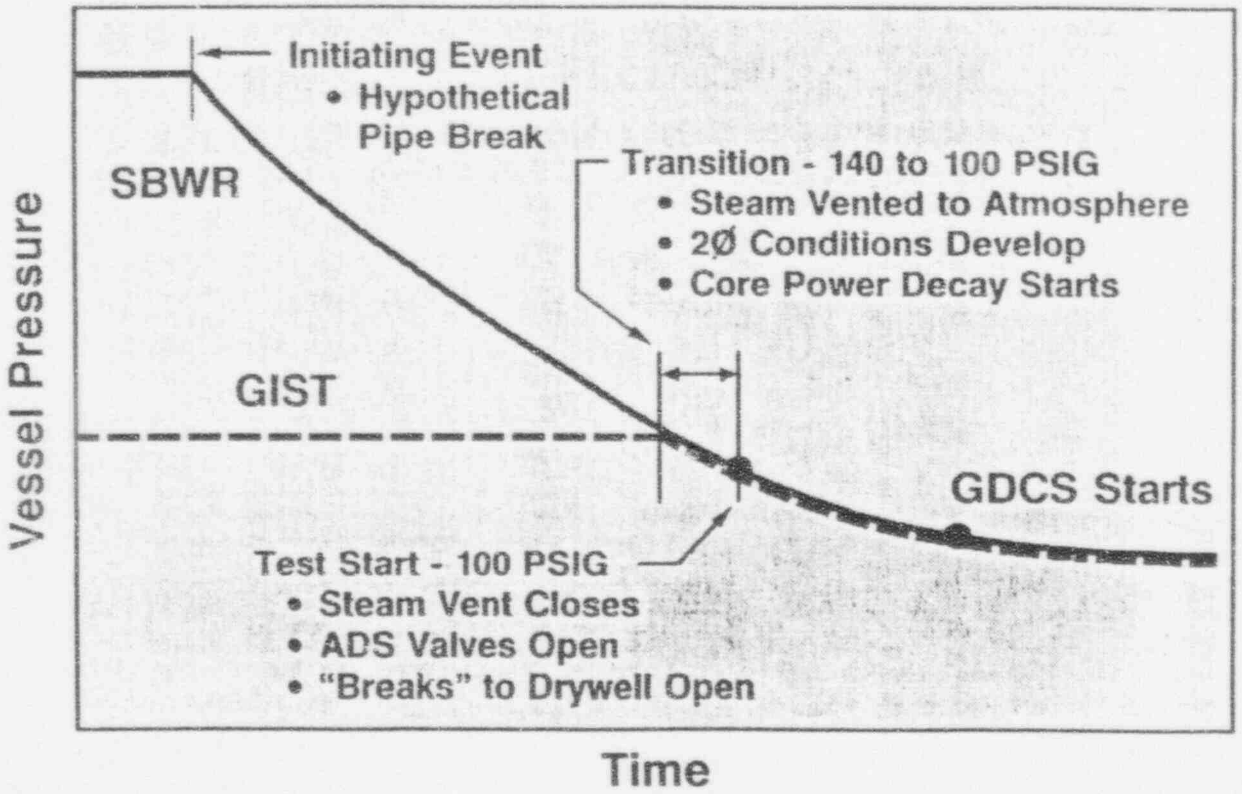
| Category | SBWR | GIST |
|---|---|---|
| **Vessel & Internals** | | |
| Downcommer geometry | Continuous Anulus | 2 separate pipes |
| Separators/dryers | Included in design | Not modeled |
| Chimney configuration | Shroud extension | Modeled with standpipes |
| Core geometry | Typical BWR | Heaters in concentric circles |
| **Depressurization System** | | |
| ADS configuration | 8 SRV piped to pool<br>6 DPV to drywell | All ADS flow to suppression pool |
| **Gravity Driven Cooling System** | | |
| Water source | Separate GDCS pools in drywell | Elevated suppressi~~ng~~on pool |
| Piping Geometry | 3 lines, each branch into 2 6" - lines | 4 - 4" lines, orificed |
| Inlet location | 1 M above TAF | 3 m above heaters |
| **Decay Heat Removal Systems** | | |
| Isolation Condenser | 3 IC HX | Not modeled |
| Passive Containment Cooling | 3 PCC HX DW to GDCS pool | Not modeled |

# GIST Operation

- Four Operator Actions
  1) Turn on computer
  2) Start test
  3) Stop test
  4) Turn off computer
- Automatic Controllers Ran the Tests
  - Put RPV pressure on depressurization "Glide Path"
  - Simulated decay heat
  - Opened DPVs
- Nature Did the Rest
  - Open GDCS check valves on low RPV pressure
  - Reflood RPV

# *RPV Glide Path*

## VESSEL PRESSURE COASTDOWN



## *GIST Matrix Tests*

*11/8 - 12/12/88*

- *21 Tests plus 3 reruns*
- *4 LOCA types studied*
  - *Main steam line break*
  - *Bottom drain line break*
  - *GDCS line break*
  - *No break ( loss of feedwater)*
- *Test Conditions*
  - *Base case*
  - *Low pool level*
  - *Increased GDCS flow*
  - *Decreased GDCS flow*

# GIST Matrix Tests (cont.)

- Test Conditions (cont.)
  - Low RPV water level
  - CRD flow
  - Increased DPV area
  - Decreased DPV area
  - High Core power
  - High wetwell pressure
  - High pool temperature
  - Low GDCS injection in RPV

# GIST Additional Tests

*12/13/88 - 12/15/88*

- *5 Tests*

- *Study phenomena and configurations not considered before*

- *Test conditions*
  - *No low pressure DPVs*
  - *Lower RPV water level - core heatup*
  - *"High" pressure supplemental injection*
  - *No core power*

Table 3.2-1

INITIAL GIST CONDITIONS

(RPV at 100 psig)

| Test(1) | No. of GDCS Lines | RPV Level (in)(2) | Scram Time (sec)(3) | Decay Heat (kW) | LDW Level (in) | UDW Press (psig) | S/P Level (ft) | S/P Temp. (°F) | WW Press (psig) |
|---|---|---|---|---|---|---|---|---|---|
| BDLB Tests: | | | | | | | | | |
| A01 Base Case | 3 | 347 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| A02 Low S/P Water Level | 3 | 347 | 369 | 89 | 4 | 13.0 | 59.2 | 105 | 6.5 |
| A03 Maximum GDCS Flow | 4 | 347 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| A04 Low RPV Water Level | 3 | 327 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| A05 CRD Flow | 3 | 347 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| A06 Minimum GDCS Flow | 1 | 347 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| A07 No Low Press DPVs | 3 | 347 | 369 | 89 | 4 | 13.0 | 67.2 | 105 | 6.5 |
| MSLB Tests: | | | | | | | | | |
| B01 Base Case | 3 | 340 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B02 Low RPV Water Level | 3 | 320 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B03 Low S/P Water Level | 3 | 340 | 212 | 99 | 6 | 14.5 | 59.2 | 110 | 7.0 |
| B04 First Repeat Test | 3 | 340 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B06 Last Repeat Test | 3 | 340 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B07 Low-low RPV WL | 3 | 300 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B08 Accumulator Makeup | 3 | 300 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| B09 Accumulator Makeup | 3 | 286 | 212 | 99 | 6 | 14.5 | 67.2 | 110 | 7.0 |
| GDLB Tests: | | | | | | | | | |
| C01A Base Case | 2 | 347 | 373 | 88 | 5 | 11.5 | 67.2 | 105 | 7.0 |
| C02 Max HP DPV Area | 2 | 347 | 373 | 88 | 5 | 11.5 | 67.2 | 105 | 7.0 |
| C03 Min HP DPV Area | 2 | 347 | 373 | 88 | 5 | 11.5 | 67.2 | 105 | 7.0 |
| C04 High LP DPV Setpt. | 2 | 347 | 373 | 88 | 5 | 11.5 | 67.2 | 105 | 7.0 |
| NB Tests: | | | | | | | | | |
| D01A Base Case | 3 | 347 | 865 | 74 | 0 | 0.0 | 67.2 | 107 | 0.0 |
| D02 Maximum GDCS Flow | 4 | 347 | 865 | 74 | 0 | 0.0 | 67.2 | 107 | 0.0 |
| D03A App. K Decay Heat | 3 | 347 | 865 | 94 | 0 | 0.0 | 67.2 | 107 | 0.0 |
| D04 Pressurized WW | 3 | 347 | 865 | 74 | 0 | 14.7 | 67.2 | 107 | 14.7 |
| D05 High Pool Temp | 3 | 347 | 865 | 74 | 0 | 0 | 67.2 | 157 | 0.0 |
| D06 Low GDCS Injection | 4 | 347 | 865 | 74 | 0 | 0 | 67.2 | 107 | 0.0 |
| D07 No Power | 3 | 347 | --- | 0 | 0 | 0 | 67.2 | 107 | 0.0 |

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

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

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

# GIST TEST RESULTS

- FEASIBILITY OF GRAVITY DRIVEN CORE COOLING SYSTEM DEMONSTRATED OVER A WIDE RANGE OF TEST CONDITIONS
- 24 UNIQUE TESTS PERFORMED
  - SUFFICIENT DATA BASE FOR TRACG QUALIFICATION OF GDCS FLOW AND RPV WATER LEVEL

# INDIVIDUAL TEST SUMMARIES

## EXCERPTS FROM GIST FINAL TEST REPORT (GEFR-00850)

### 3.4.1 Bottom Drain Line Break

Table 3.4-1 presents the results of the BDLB tests.

#### 3.4.1.1 Base Case (Test A01)

The BDLB tests are characterized by a relatively slow blowdown rate for a LOCA. The break being at the bottom of the vessel does not contribute to the RPV depressurization. For the base case (Test A01), three GDCS lines provided makeup flow to the vessel.

3.4.1.2 Low S/P Water Level (Test A02)

In Test A02 the S/P water level was lowered eight feet, effectively lowering the driving head for the GDCS by the same amount. While this delayed the start of GDCS flow for 27 seconds, the final minimum water level in the RPV remained about the same because the collapse of voids in the lower plenum occurred later.

3.4.1.3 Maximum GDCS Flow (Test A03)

In Test A03 all four GDCS lines provided flow to the vessel. While the minimum water level in the RPV occurred sooner, the level itself was the same as the base case. The additional cold water caused a faster RPV depressurization; however, when the minimum level occurred, the same amount of water had been added as that of the base case. This test indicated that there is considerable margin in the required GDCS flow.

3.4.1.4 Low RPV Water Level (Test A04)

The initial RPV water level was 20 inches lower in this test. As expected, the blowdown rate was not greatly affected, but the final minimum water level was lower.

3.4.1.5 CRD Flow (Test A05)

A small flow of water was injected into the bottom of the RPV throughout this test to represent the control rod drive (CRD) flow that occurs in the SBWR. The voids in the lower plenum collapsed sooner, caused by this flow cooling the water there; however, the minimum water level recorded was higher because of this additional inventory. This indicates that the assumed failure of CRD flow in accident analyses is conservative.

3.4.1.6 Minimum GDCS Flow (Test A06)

Only one GDCS line was used in this test. The minimum RPV water level was lower, as expected. Since the injection flow was only slightly higher than the break flow, reflood was extremely slow. This was the only test that was halted before the vessel had refilled above the core level.

3.4.1.7 No Low Pressure DPVs (Test A07)

The low pressure DPVs remained closed during this test. The depressurization rate was, therefore, slower. However, this also had the effect of delaying the void collapse in the lower plenum even more than the delay to start GDCS injection. Additional GDCS makeup caused a higher minimum water level in the RPV.

Test A07 was one of the five tests chosen for TRACG qualification. This test was chosen because the final SBWR conceptual design does not include low pressure DPVs (see Appendix), and this test most closely represents the ADS capacity of that design.

3.4.2 Main Steam Line Break

Table 3.4-2 presents the results of the MSLB tests.

3.4.2.1 Base Case (Test B01)

The MSLB tests were characterized by the most rapid blowdown of the RPV. Because of this rapid blowdown and inventory loss, this base case was used to test the repeatability of GIST by rerunning it later in the program. These "repeatability" runs (Tests B04 and B06) are discussed below. Test B01 was selected as one of the five tests used to qualify TRACG, since it represents the most limiting LOCA condition for SBWR.

3.4.2.2 Low RPV Water Level (Test B02)

In Test B02 the initial RPV water level was 20 inches lower than in the base case. The subsequent minimum level recorded during the test was barely over the core. The core stayed covered throughout the test and did not heat up.

3.4.2.3 Low S/P Water Level (Test B03)

In this test the initial S/P water level was lowered by eight feet. This variation on the base case was identical to that of Test A02 (Section 3.4.1.2) as were the results.

3.4.2.4 First Repeatability Run (Test B04)

This test repeated the conditions of the base case (Test B01). The results were the same. Since no drift was seen in the GIST instrumentation, GE decided to cancel the second repeatability test that was part of the test matrix (Test B05) in order to fit in additional special tests.

3.4.2.5 Last Repeatability Run (Test B06)

As with Test B04, this test repeated the conditions of the base case (Test B01). The test was run at the conclusion of the original test matrix. The results were also the same. Additional tests (B07, B08, B09, A07, D07, and D01A) not on the matrix were run after this test before shutting down the GIST Facility.

3.4.2.6 Low-low RPV Water Level (Test B07)

In this test the initial RPV water level was 40 inches below that of the base case. When the voids collapsed in the lower plenum, the core uncovered and started to heat up. The GIST emergency safety function kicked in and cut off power to the core heater rods. The purpose of this test was to confirm that there was a point below which the core would uncover and heat up.

Test B07 was one of the five tests chosen for TRACG qualification. This test was chosen to challenge TRACG to accurately predict that the core will uncover given this initial water inventory.

3.4.2.7 Accumulator Makeup (Tests B08 and B09)

Two tests were run examining the behavior of having a high pressure accumulator supply makeup water in addition to the GDCS. Both tests had 93 pounds of water injected into the RPV at 80 psig. The injection lasted about 200 seconds. Test B08 was a repeat of Test B07 initial conditions with the low-low initial RPV water level. Test B09 had an even lower initial water level--54 inches below the base case. In both tests, the core never uncovered.

The final SBWR conceptual design does not include high pressure accumulator makeup systems. These tests were added at the end of the test program when there was still discussion of including them in the final design. Since these tests are not representative of the final SBWR design, they were not used in the TRACG qualification effort.

3.4.3 GDCS Line Break

Table 3.4-3 presents the results of the GDLB tests.

3.4.3.1 Base Case (Tests C01 and C01A)

The GDLB tests were characterized by having the slowest reflood time of all the tests. With one GDCS line unavailable because of the break and another out by the assumed single failure, only two GDCS lines were used for most of these tests.

Base case Test C01 was repeated in Test C01A because the vacuum breakers between the wetwell (WW) and upper drywell (UDW) were not functioning. The broken GDCS line injected not only hot water and steam from the vessel but also cold water from the S/P. This cold water condensed the steam in the UDW, lowering its pressure below that of the WW. Since the vacuum breakers were out of service in Test C01, there was a danger of back-flow from the S/P through the main vents. This could have caused such a rapid (and non-SBWR typical) depressurization in the UDW that the GIST Facility could have been damaged. To prevent this, the operator continually injected steam into the UDW during the test. Since this injection was not measured and the containment behavior was not typical of the SBWR (as all base cases were required to be), the test was invalid. The vacuum breakers were functional in Test C01A and all other GDLB tests.

Test C01A was one of the five tests chosen for TRACG qualification. This test was chosen to challenge TRACG to accurately predict the depressurization of the UDW and the operation of the wetwell-to-drywell vacuum breakers.

The GDLB tests were used to study variations in the DPV pressure setpoints. The results generally showed that there is considerable margin in these setpoints. Small changes have little effect on the final result of minimum RPV water level.

#### 3.4.3.2 Maximum High Pressure DPV Area (Test C02)

In this test the DPV flow area of the high pressure valves was increased 50%, while the flow area of the low pressure valves was decreased 50%. The resultant total flow area remained the same. This had the effect of modeling a change in the pressure setpoints of one-fourth of the SBWR DPVs. The RPV depressurized faster, allowing the GDCS flow to begin sooner; however, the minimum water level recorded was about the same.

#### 3.4.3.3 Minimum High Pressure DPV Area (Test C03)

This test was the opposite of Test C02 in that the DPV flow area of the high pressure valves was decreased 50%, while the flow area of the low pressure valves was increased 50%. The RPV depressurized slower, causing the

GDCS flow to begin later. However, the void collapse in the lower plenum was also later, resulting in approximately the same minimum water level.

3.4.3.4 Higher Low Pressure DPV Setpoint (Test C04)

In this test the low pressure DPVs were activated at 90 psig instead of the usual 60 psig. While the minimum water level in the RPV was lower than the base case, the difference was not significant.

3.4.4 <u>No break</u>

Table 3.4-4 presents the results of the no-break (NB) tests.

3.4.3.1 Base Case (Tests D01 and D01A)

The NB tests were characterized by having the containment remain at atmospheric pressure. In the early SBWR conceptual design, the ADS was piped to the S/P. Without a LOCA, there was no reason for the containment to pressurize. The final SBWR conceptual design has the ADS discharging into the UDW, thereby causing the containment to pressurize even without there being a pipe break (see the Appendix for a complete description of both ADS designs).

The power did not decay as expected in Test D01 so a repeat was necessary (Test D01A). The difference in decay heat was so slight that the two runs are almost identical.

3.4.3.2 Maximum GDCS Flow (Test D02)

In this test all four GDCS lines were operational, but half of the low pressure DPVs were assumed inoperative. The failed DPVs were modeled in GIST by reducing the flow area of the two low pressure DPVs by one half. This represented a failure of six of the twelve SBWR DPVs and effectively cut the total ADS area by one quarter. That was greater than a single valve failure even on the final conceptual SBWR design, which contains six DPVs and four dual function safety-relief valves. The minimum RPV water level was raised ten inches above the base case.

3.4.3.3 Appendix K Decay Heat (Tests D03 and D03A)

These tests used the more conservative Appendix K decay heat tables to model the decay heat of the core. As expected, a higher heating rate will result in a lower minimum water level. Test D03 was invalid, since the power going to the heater coils was too low. In Test D03A this power was corrected.

Test D03A was one of the five tests chosen for TRACG qualification. This test was chosen to challenge TRACG to accurately predict the behavior of GIST with the Appendix K decay heat.

3.4.3.4 Pressurized Containment (Test D04)

In this test, the containment is pressurized to two atmospheres before the start of the test. In this way, the test modeled the behavior of a no-break accident where the ADS is routed to the containment. As expected, the increased head on the GDCS pool allowed the GDCS to initiate sooner and resulted in a higher minimum RPV water level. This test proves that the design change in the final conceptual SBWR design to have the ADS discharge to the UDW is an improvement over the earlier design.

3.4.3.5 High Pool Temperature (Test D05)

The S/P was approximately 50°F hotter in this test. The results were not significantly different than the base case. This test shows that there is a large margin in GDCS makeup temperature which has little effect on the results.

### 3.4.3.6 Low GDCS Injection (Test D06)

This test was a repeat of Test D02 (Maximum GDCS Flow), except two of the GDCS lines injected water near the bottom of the annulus. The purpose of this test was to see if injecting cold water low in the vessel to minimize steam quenching would cause collapsing of the voids sooner and a lower minimum water level. However, in GIST the lower injection point increased the head on those lines and caused them to inject sooner. This, in turn, increased the total makeup before void collapse and thus raised the RPV minimum water level.

### 3.4.3.7 No Power (Test D07)

This test, one of the last added at the end of the testing program, was simply a sensitivity study on RPV behavior when no heat is added to the core. It provides an additional data point (with Test D03A) on the effect of decay heat alone on the results. Not surprisingly, the absence of decay heat is beneficial to reflooding the vessel.

Table 3.4-1

BDLB TEST RESULTS

| Test | | Date | Test Times from RPV at 100 psig (sec) to 60 psig | to GDCS Inj. | to Min. WL | to Test Stop | Annulus Water Levels (in) Init. | Min. | Frothy Annulus |
|---|---|---|---|---|---|---|---|---|---|
| A01 | Base Case | 11-29-1988 | 119 | 285 | 423 | 680 | 343 | 256 | B |
| A02 | Low S/P Water Level | 11-30-1988 | 120 | 312 | 470 | 737 | 344 | 255 | B |
| A03 | Maximum GDCS Flow | 12-01-1988 | 124 | 293 | 408 | 599 | 343 | 256 | B |
| A04 | Low RPV Water Level | 12-02-1988 | 116 | 277 | 412 | 768 | 325 | 242 | A |
| A05 | CRD Flow | 12-06-1988 | 113 | 282 | 403 | 609 | 343 | 269 | B |
| A06 | Minimum GDCS Flow | 12-01-1988 | 122 | 306 | 445 | 924 | 343 | 246 | A |
| A07 | No Low Press DPVs | 12-14-1988 | 127 | 489 | 674 | 866 | 344 | 267 | A |

Table 3.4-2

MSLB TEST RESULTS

| Test | | Date | Test Times from RPV at 100 psig (sec) to 60 psig | to GDCS Inj. | to Min. WL | to Test Stop | Annulus Water Levels (in) Init. | Min. | Frothy Annulus |
|---|---|---|---|---|---|---|---|---|---|
| B01 | Base Case | 11-08-1988 | 79 | 215 | 320 | 648 | 338 | 238 | A |
| B02 | Low RPV Water Level | 11-09-1988 | 76 | 200 | 301 | 691 | 316 | 226 | A |
| B03 | Low S/P Water Level | 11-10-1988 | 82 | 237 | 344 | 705 | 337 | 235 | A |
| B04 | First Repeat Test | 11-28-1988 | 80 | 209 | 313 | 907 | 338 | 240 | A |
| B06 | Last Repeat Test | 12-12-1988 | 87 | 220 | 321 | 667 | 338 | 241 | B |
| B07 | Low-Low RPV WL | 12-12-1988 | 68 | 188 | 301 | 1169 | 295 | 188 | A |
| B08 | Accumulator Makeup | 12-13-1988 | 66 | 204 | 239 | 1124 | 295 | 232 | A |
| B09 | Accumulator Makeup | 12-13-1988 | 64 | 170 | 234 | 644 | 286 | 239 | A |

Table 3.4-3
GDLB TEST RESULTS

| Test | Date | Test Times from RPV at 100 psig (sec) to 60 psig | to GDCS Inj. | to Min. WL | to Test Stop | Annulus Water Levels (in) Init. | Min. | Frothy Annulus |
|---|---|---|---|---|---|---|---|---|
| C01 Base Case | 12-07-1988 | 118 | 290 | 464 | 1168 | 343 | 254 | B |
| C01A Base Case | 12-08-1988 | 123 | 295 | 435 | 971 | 343 | 253 | A |
| C02 Max HP DPV Area | 12-09-1988 | 98 | 259 | 469 | 917 | 343 | 258 | B |
| C03 Min HP DPV Area | 12-09-1988 | 186 | 382 | 535 | 1019 | 344 | 256 | B |
| C04 High LP DPV Setpt. | 12-08-1988 | 86 | 264 | 429 | 1210 | 343 | 243 | A |

Table 3.4-4

NB TFST RESULTS

| Test | | Date | Test Times from RPV at 100 psig (sec) to 60 psig | to GDCS Inj. | to Min. WL | to Test Stop | Annulus Water Levels (in) Init. | Min. | Frothy Annulus |
|---|---|---|---|---|---|---|---|---|---|
| D01 | Base Case | 11-11-1988 | 118 | 333 | 497 | 760 | 344 | 253 | A |
| D01A | Base Case | 12-16-1988 | 122 | 343 | 499 | 782 | 344 | 253 | A |
| D02 | Maximum GDCS Flow | 11-15-1988 | 120 | 395 | 562 | 720 | 344 | 263 | A |
| D03 | App. K Decay Heat | 11-16-1988 | 126 | 347 | 501 | 783 | 344 | 251 | B |
| D03A | App. K Decay Heat | 11-17-1988 | 125 | 361 | 517 | 816 | 344 | 244 | B |
| D04 | Pressurized WW | 11-18-1988 | 120 | 244 | 353 | 531 | 344 | 269 | A |
| D05 | High Pool Temp. | 11-18-1988 | 119 | 338 | 481 | 773 | 345 | 257 | A |
| D06 | Low GDCS Injection | 11-21-1988 | 118 | 398 | 512 | 708 | 344 | 269 | A |
| D07 | No Power | 12-15-1988 | 97 | 274 | 368 | 569 | 343 | 283 | A |

# ATTACHMENT 6

# PANDA TEST

CONTAINS PAPER BY PAUL CODDINGTON ET. AL. ON PSI ALPHA PROGRAM

# ALPHA — THE LONG-TERM PASSIVE DECAY HEAT REMOVAL AND AEROSOL RETENTION PROGRAMME

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## Abstract

The Paul Scherrer Institute has recently initiated the major new experimental and analytical programme ALPHA, which is aimed at understanding the long-term decay heat removal and aerosol questions for the next generation of Passive Light Water Reactors. The ALPHA project currently includes four major items: the large-scale, integral system behaviour test facility PANDA, which will be used to examine multidimensional effects of the SBWR decay heat removal system; an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes (LINX); a separate-effects study of aerosol transport and deposition in plena and tubes (AIDA); while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of relevant system codes. The paper briefly reviews the above four topics, and discusses some aspects of the criteria and scaling used to guide the design of the PANDA experimental facility.

## 1 Introduction

The Paul Scherrer Institute has recently initiated the major new experimental and analytical programme ALPHA (Advanced Light Water Reactor Passive Heat Removal and Aerosol Retention Programme), which is aimed at understanding the long-term decay heat removal and aerosol questions for the next generation of Passive Light Water Reactors. The ALPHA project currently includes four major items: the large-scale, integral system behaviour test facility PANDA (Passive Nachwaermeabfuhr und Druckabbau Testanlage); an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes (LINX); a separate-effects study of aerosol transport and deposition in plena and tubes (AIDA); while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of the relevant system codes.

## 2 PANDA - An integral Containment Simulation Facility

### 2.1 Introduction

A good understanding of the behaviour of the relatively novel containment concepts proposed for the future advanced passive LWRs is of importance when assessing their safety. These concepts rely on natural circulation cooling modes; their long-term behaviour includes the mixing of steam and non-condensible gases, condensation of such mixtures in parallel condenser units, large open tanks and water pools, and the mixing of fluids in large pools, air volumes, etc. Integral containment system behaviour may exhibit multi-dimensional effects, due, for example, to incomplete mixing and varying modes of operation of parallel units. The PANDA facility has been designed to address such questions at a relatively large scale.

The PANDA facility will consist of a 1.5 MW steam source and a number of large pressure vessels, typically 4 m in diameter and 8 m high, which can be interconnected by external piping and may contain internal structures, representing the various compartments of a variety of reactor containments. The vessels will be fitted with instrumentation to measure fluid temperatures, levels, pressures and flows as well as steam and gas concentrations.

In the first instance, the PANDA facility will be used to examine multidimensional effects for the General Electric Simplified Boiling Water Reactor (SBWR) decay heat removal system. The SBWR utilizes two types of condenser units (Fig. 1) to remove the reactor decay heat, following a Loss-Of-Coolant Accident, from the reactor containment to an outside water pool. First, there are three Isolation Condensers (IC) connected to the reactor primary system, which will be used to remove the decay heat during a reactor isolation at full pressure. The PANDA facility will include scaled models of these units to investigate their behaviour during an accident; it will not, however, simulate their high pressure, reactor isolation, decay heat removal function. Second, there are, currently, for the SBWR and PANDA, two low-pressure condenser units connected directly to the reactor containment (Drywell), referred to as Passive Containment Coolers or PCC units. The experimental facility PANDA will examine, on a large scale (1/25 volumetric), the system interactions between the multiple condenser units, and their heat removal capacity in the presence of non-condensible gases such as nitrogen and helium (as a simulant of hydrogen). The PANDA system behaviour tests will extend the data base of previously performed experiments [1] to a much larger scale, study the interaction between the various PCC and IC units, and provide verfication of integral system behaviour under a variety of conditions.

The PANDA simulation of the SBWR (Fig. 2) will consist of a representation of the reactor pressure vessel (RPV), reactor containment (Drywell) and suppression pool (Wetwell), as well as the Isolation Condenser and Passive Containment Cooler units and their associated water pools. Finally, condensate will be collected in a "condensate catch tank" simulating the Gravity Driven Cooling System (GDCS) pool in the SBWR.

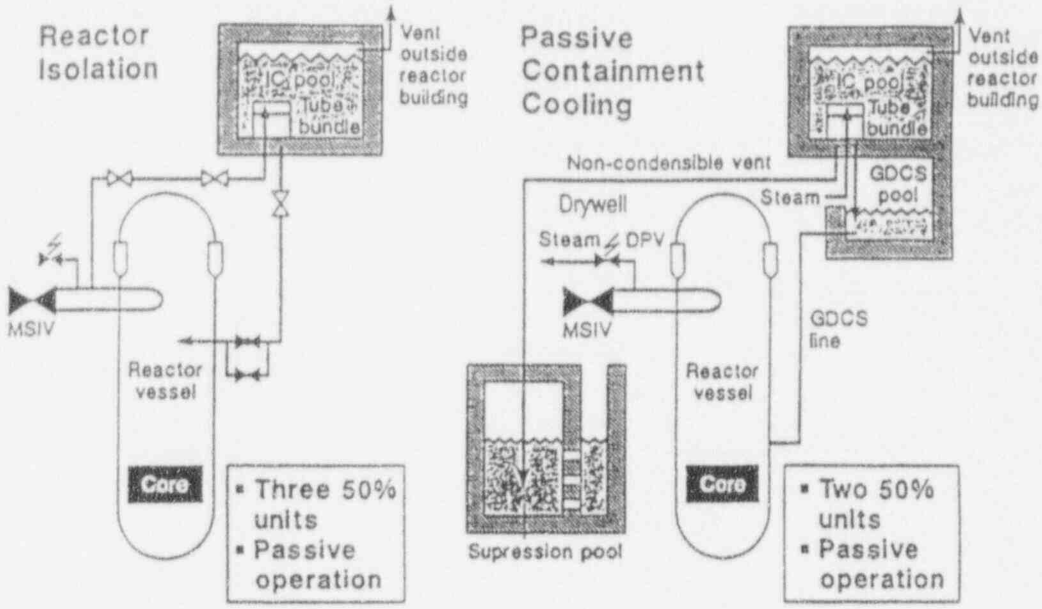


Figure 1: SBWR Isolation Condensers and Passive Containment Coolers.

## 2.2 General Guidelines

Early during the conceptual design phase of the facility, it was recognized that it is neither possible nor desirable to preserve exact geometrical similarity between the reactor containment volumes and the experimental facility. On the other hand, multidimensional containment phenomena such as mixing of gases and natural circulation between compartments may depend on the particular geometry of the containment building. The general philosophy followed in designing the experimental facility was to allow such multi-dimensional effects to take place by dividing the main containment compartments in two and by providing a variety of well-controlled boundary conditions (e.g. imbalances) during the experiments, so that the various phenomena could be studied parametrically under well-established conditions, and a behaviour envelope of the system established. Carefully conducted parametric experiments can also provide more valuable data for code validation than attempts to simulate geometrically, but to an insufficient degree, the rather complex reactor system. Boundary conditions and the behaviour of the interconnections between the various containment volumes can be controlled externally by software to study various system scenarios and alternative accident paths.

Beyond the general considerations stated above, in designing the PANDA facility and, in particular, the main vessels, the following general guide lines were followed:

- Full vertical height should be preserved, to correctly represent the various gravity head driving forces.
- The system should be modular and use simple interconnected cylindrical vessels to simulate possible 3-dimensional effects in the SBWR annular geometry.
- Volumes should be minimised to the extent compatible with the preservation of the scaling factor chosen and the system behaviour.
- The power-to-volume scaling ratio should be preserved and should be as large as practically possible.
- The experiments will be conducted under reactor pressure and temperature conditions. (The facility is designed for nominal operation at 10 bar and 180 °C).

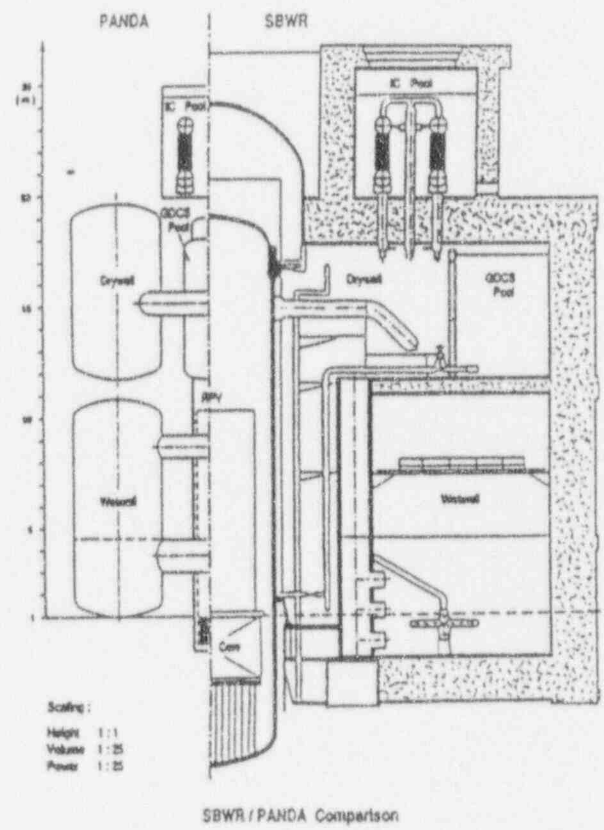


Figure 2: SBWR Containment and PANDA Comparison.

Figure 3 shows the current geometrical arrangement of the proposed PANDA facility with two interconnected Drywells, two interconnected Wetwells, the reactor pressure vessel (RPV), and a tank (GDCS Pool) to catch the steam condensate prior to returning it to the RPV. It was decided to represent the SBWR Drywell and Wetwell with two units in the PANDA facility, in order to better examine, in a systematic manner, the possible spatially non-uniform mixture of nitrogen and steam flowing through the condenser, IC and PCC units. It was considered necessary to be able to investigate the venting and purging of each of the condenser units for different mixtures of nitrogen and steam flowing into each unit, and also to examine the energy deposition and distribution in the Wetwell pool, resulting from the venting of uncondensed steam, under such asymmetric conditions. The volumetric scaling of the PANDA facility shown in Fig. 3 is 1/25. Figure 2 shows the elevations of PANDA relative to those of the SBWR containment. All the SBWR heights are represented except those below the top of the active fuel (TAF). The argument for reducing the facility height by eliminating the fluid below the TAF was that this liquid is essentially inactive and is not required to correctly simulate the gravity heads, or, in the instance of the Wetwell, likely to be involved in the absorption of any energy passed into the Wetwell in the form of uncondensed steam. Therefore, for a given facility budget it was considered preferable to eliminate this volume from PANDA and so increase the overall scale of the facility. Eliminating dead volumes also decreases preconditioning times and fluid inventories and increases experimental flexibility.

## 2.3 Scaling: The IC and PCC Condenser Units

The 1:25 scaling factor chosen for the PANDA facility is, of course, a compromise between several factors. On the one hand there is the requirement to keep the PANDA vessels within a manageable size and cost, while at the same time the desire is to construct as large a facility as possible, to provide a meaningful basis for extrapolation from the previous 1:400 scale Isolation Condenser decay heat removal experiments [1] to the full reactor scale. A critical factor that led to the choice of a 1:25 scale was the requirement that the condenser unit secondary side behaviour should be representative of the units to be used in the SBWR. Figure 4 provides a schematic of a condenser module; there are two such modules per condenser unit in the SBWR (see [2] for more details). We can also see in Fig. 4 how it is possible to construct a unit at the PANDA scale by taking a slice from an SBWR condenser. Having made the decision to fabricate the PANDA condensers from a slice of the SBWR units, the only question then is how wide this should be, and Fig. 4 and Table 1 show how a 3-tube-wide slice corresponds to a scale of 1:25. This is the minimum width that will permit some tubes to be totally surrounded by other tubes. In all other respects (height, pitch, diameter, and wall thickness) the PANDA condenser tubes are identical to those to be used in the SBWR.

From Table 1, we see that adopting the above procedure

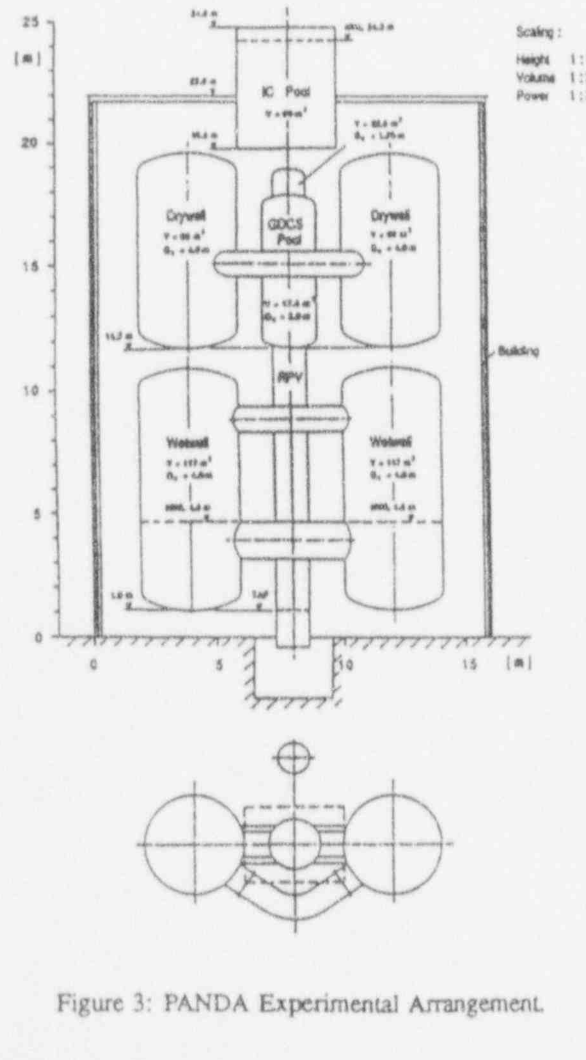


Figure 3: PANDA Experimental Arrangement.

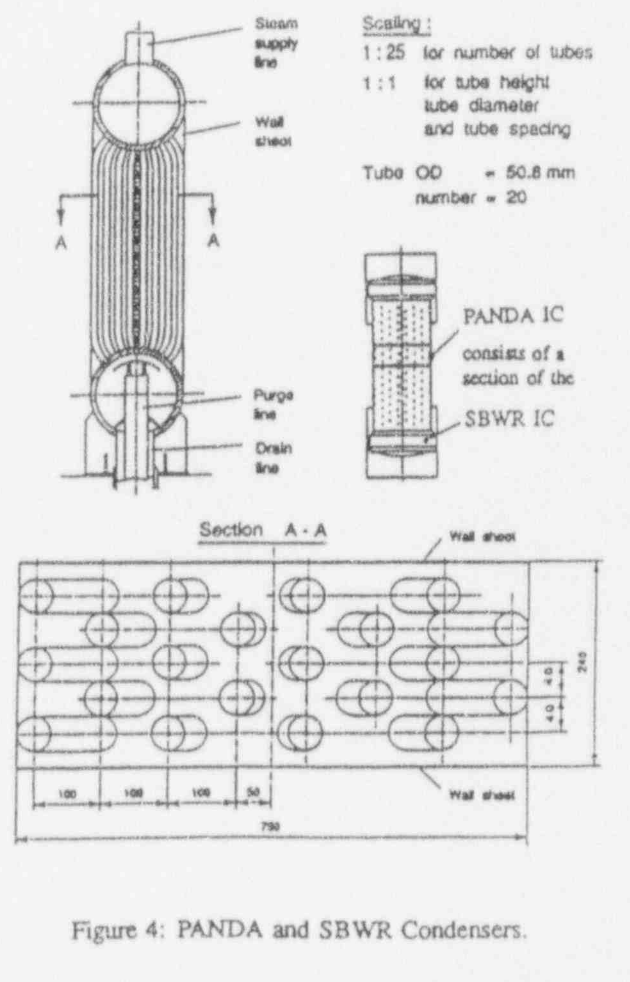


Figure 4: PANDA and SBWR Condensers.

Table 1: PANDA and SBWR, IC and PCC Scaling.

| Low pressure PCC | High pressure IC |
|---|---|
| SBWR:<br>Number of units: 2 (2 x 50 %)<br>tubes: 496 (per unit) | SBWR:<br>Number of units: 3 (3 x 50 %)<br>tubes: 240 (per unit)<br>480 (for 2 units = 100%) |
| PANDA:<br>Number of scaled units: 2 (2 x 50%)<br>Scaling Factors:<br>1:25 for number of tubes<br>(496/25 = 19.84 → 20 per unit)<br>1:1 for tube height, pitch and diameter | PANDA:<br>Number of scaled units: 1 (1 x 100%)<br>Scaling Factors:<br>1:25 for number of tubes<br>(480/25 = 19.2 → 20 per unit)<br>1:1 for tube height, pitch and diameter |

for the IC produces a single unit in PANDA that has twice the tube area, at the 1:25 scale, of that of an SBWR IC.

This means that the PANDA facility will have three condenser units, two equivalent to the two SBWR PCCs and one equivalent to two SBWR ICs.

## 2.4 The PANDA Vessels and Power Source

A schematic of the PANDA vessels is given in Fig. 3 while an isometric view is shown in Fig. 5.

As a example of the application of the general guidelines stated above, as well as of other secondary considerations, the design of the PANDA Wetwell vessel is outlined as follows:

- In order to preserve the pressure response of the entrapped non-condensible gas, it is necessary to scale the net Wetwell vapour space.
- To have a correct representation of the evaporation/ condensation processes at the pool surface, it is necessary to correctly scale the total Wetwell pool surface area.
- To provide a representative volume of water with which the uncondensed steam vented into the suppression pool can mix; the water pool depth must extend sufficiently below the condenser vent line. The suppression pool depth was also required to be large enough to accommodate at least the topmost main (horizontal) vent and the Wetwell-to-RPV equalisation line. This was, in fact, the limiting factor in determining the pool depth.

In this manner it was possible to define the Wetwell dimensions. Similar procedures were also used to define the Reactor Pressure Vessel (RPV) and Drywell. In the case of the Drywell, the most important parameter to scale (for a well-mixed system) is the total volume, since this and the power level determine the venting time of the Drywell nitrogen to the PCC units.

The lower part of the Drywell volume surrounding the RPV was not included in the height of the PANDA Drywell volume, since it was felt that possible natural circulation phenomena taking place in this annular volume (heated on one side by the RPV) could not be adequately modelled. The volume of the annular space was, however, included in the PANDA Drywell volume.

For ease of construction it was considered desirable to have the Drywell and Wetwell tanks of the same diameter. Not all processes, and in particular the detailed mixing of the nitrogen and the steam from the RPV in the Drywell and the mixing of the uncondensed steam with the suppression pool water, can be accurately simulated in a scaled facility such as PANDA. In these instances separate effects studies, both experimental and analytical (see Section 4), will be used to guide parametric studies in the PANDA facility.

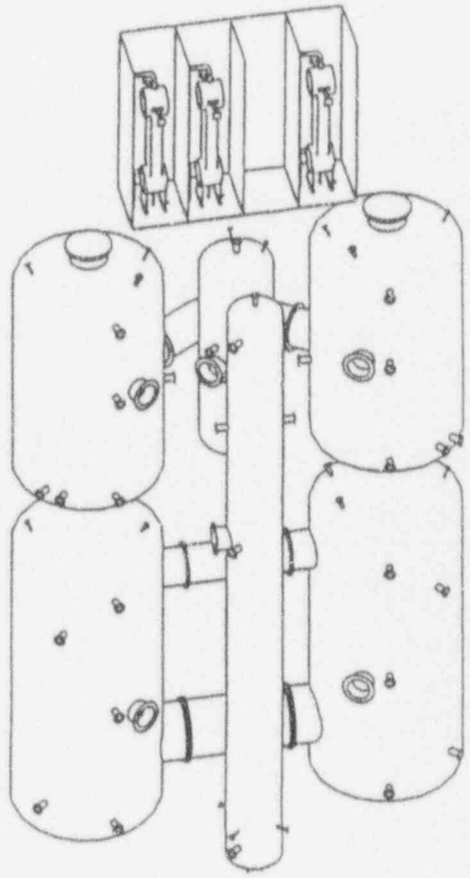

Figure 5: Isometric View of PANDA Vessels.

For example, nitrogen may be injected into the Drywell to simulate the slow convection of trapped nitrogen from a compartment with a restricted connection to the main Drywell.

The last two vessels shown in Figs. 2, 3 and 5, are those of the condensate catch tank (labelled GDCS pool) and the IC/PCC water pool. The requirements for these two vessels are somewhat different from those of the RPV, Drywell, and Wetwell. For example, for the PANDA IC/PCC water pool, in addition to providing sufficient water to keep the condenser tubes covered for a reasonable time (say 24 hours), the main requirement was one of flexibility. An element of the design was the requirement that the IC/PCC units could be re-configured in as many ways as possible, to follow possible changes in the SBWR design, without major impact on the programme cost and/or time schedule. Also, there was a requirement to have the capability of re-filling the pool, during the course of an experiment, with water at different temperatures, in order to examine a variety of possible SBWR long-term depressurisation strategies. As can be seen from Fig. 5, the IC/PCC pool has four inter-connected compartments and will be placed on the roof of the PANDA building (Fig. 3).

The power to the PANDA facility will be provided by electrical heaters placed near the bottom of the RPV (Fig. 6). The heaters are not designed to represent the reactor core, but will be placed so that their tops have the same relative elevation as the top of the active fuel (TAF) in the SBWR. The power level required for PANDA was determined on the basis that a PANDA transient would be initiated after reactor blowdown and follow the emptying of the GDCS water into the RPV. These events are predicted to occur within one hour of accident initiation and reactor scram, and so the required PANDA power level was set to be equal to the scaled decay heat one hour after scram. For a 1800 MW reactor, the decay heat after one hour is approx. 24 MW and so, for PANDA, approx. 1 MW of power is required. In order to provide flexibility of operation, the PANDA heaters will have a maximum installed capacity of 1.5 MW. A controller will be provided to follow accurately any given decay heat curve.

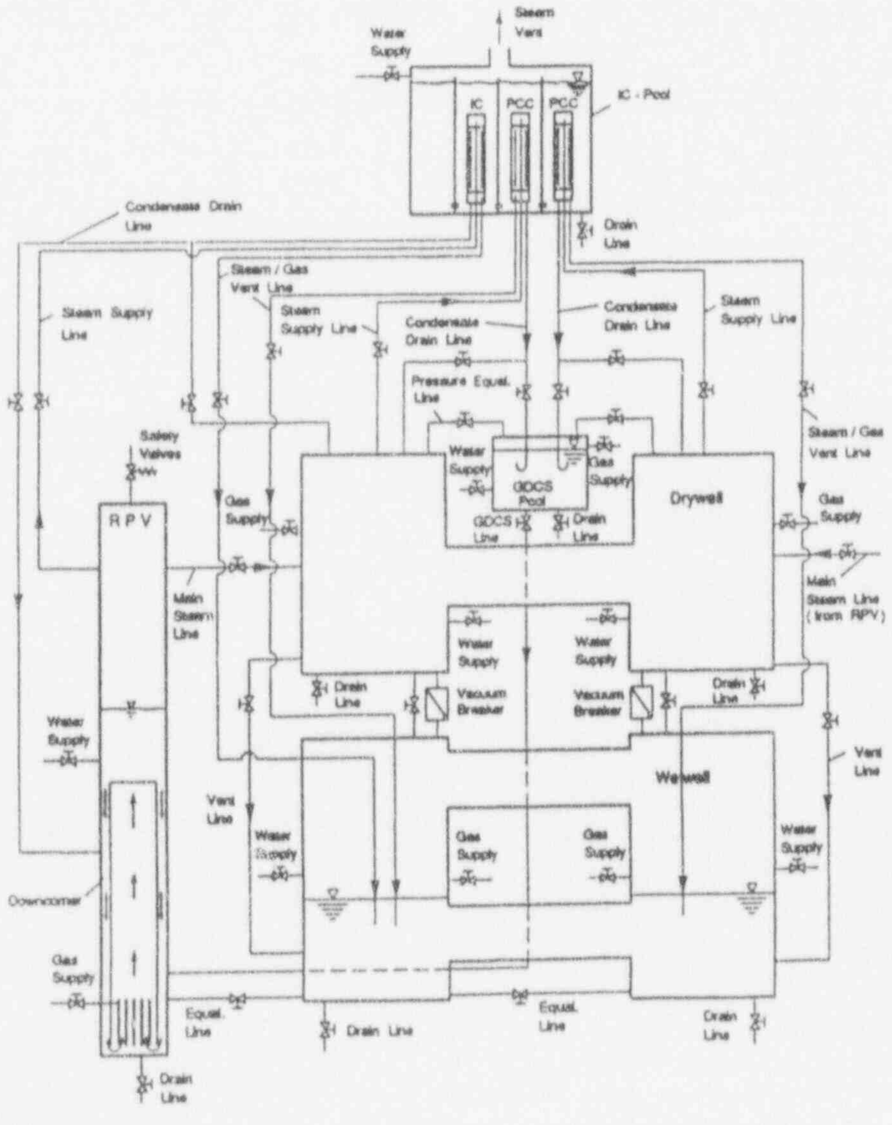


Figure 6: PANDA schematic including piping configuration.

## 2.5 Valves, Piping, and other components

The piping configuration of the PANDA facility is shown in Fig. 6, and a number of features of the design are worthy of explanation.

- All the lines (pipes) will be valved to provide maximum flexibility and ease of re-configuring the system with minimum cost and time delay.

- The schematic (Fig. 6) shows the steam line, drain line and vent line to each of the condenser units, and the PANDA simulation of the main (horizontal) vents. The main vents will not be fully scaled, since they are not predicted to clear during the course of a PANDA transient due to the small Drywell to Wetwell pressure drop, which results from the fact that the PANDA transients are not initiated until one hour after scram.

- Also shown are two vacuum breakers, each one connecting one of the two Drywell-Wetwell vessel combinations. The vacuum breakers are predicted [3] to have a major influence on the behaviour of the PANDA facility and are therefore a critical element in both the design of the SBWR containment and PANDA. Programmable control valves will therefore be used in PANDA to simulate the SBWR vacuum breakers; this will allow a variety of SBWR vacuum breaker designs to be tested with only software, rather than hardware, changes.

Finally, Fig. 6 shows the water and gas supply lines that will be used to initialise any given PANDA experiment. Sufficient flexibility will be built into the facility to investigate the effect on the transient behaviour of, for example:

- A variety of suppression pool water temperature distributions, eg. well mixed, stratified, ...

- Water pools in the Drywell to simulate liquid line breaks, eg GDCS or IC return line breaks.

- A variety of IC/PCC water pool temperature distributions.

## 2.6 Heat Losses and Heat Capacities

Major factors that can influence the behaviour of a small scale test facility, in comparison to the reactor, are the relative magnitude of heat losses and system heat capacities. In a wide range of integral test facilities it has been necessary to go to significant sophistication, including the use of guard heaters, to reduce heat losses to an acceptable level. In general, heat losses increase in inverse proportion to the scale of the facility, as the surface area to volume ratio increases at smaller scales. In this respect, PANDA at 1:25 scale is in a relatively good position. In addition, test facilities may have extra heat losses associated with additional valves, instrument penetrations, etc. Two design goals have been set for the PANDA facility:

- The heat losses, at all times during any transient, should be less than 10% of the prevailing decay heat level. Initial estimates indicate that this is achievable using commercially available insulation and that guard heaters will not be required.

- All the piping, RPV, Drywell, Wetwell, etc. should be capable of being configured to separately estimate their individual heat losses, for the range of power levels expected during the course of a transient.

Finally, the heat capacity of a steel-lined reinforced concrete structure, such as a reactor containment, cannot be simulated by a steel pressure vessel. It is anticipated, therefore, that additional material with well-defined mass, material, and geometry will be introduced into the PANDA Drywell and Wetwell to provide the required heat capacity.

This will permit the study, in a systematic and controlled manner, of the influence of heat capacity on the course of a PANDA transient.

# 3 Analytical Methods

In order to relate the design and potential behaviour of the PANDA facility to the SBWR containment when subject to a Loss-Of-Coolant Accident, the TRACG code is being used to simulate both the SBWR containment and PANDA. Figure 7, for example, shows a schematic of the TRACG representation of the PANDA facility, while Fig. 8 shows the predicted Drywell and Wetwell nitrogen and total pressures during the first 20 hours of a typical transient. The boundary conditions of the transient presented in Fig. 8 are that all the steam from the RPV is directed towards one of the two Drywells (i.e. DW1), and there is no steam flow into the PANDA IC. Figure 8 shows that the nitrogen remaining in the Drywell is vented into the Wetwell in the first 5,000 s, while the maximum Drywell pressure is not reached until about 20,000 s, when the heat removal rate of the condensers equals the decay heat generation rate. Following this, the Drywell pressure can be seen to "oscillate" about the Wetwell pressure; bounded on the low side by the opening of the vacuum breakers and the return of nitrogen to the Drywell, and on the upper side by the venting of nitrogen and steam from the Drywell to the Wetwell through the PCC units. This cyclic behaviour ensures that the "time averaged" heat removal rate of the PCC units matches the decreasing decay heat generation rate. Further details of the TRACG calculated transients for PANDA can be found in [3].

# 4 The LINX Programme

In support of the large-scale integral system behaviour PANDA tests, an investigation of natural circulation and mixing phenomena in single- and multi-phase/multicomponent systems in large pools will be conducted. This work will rely heavily on the application of Computational Fluid Dynamics (CFD) tools adapted for multiphase flow and verified against a range of both large- and small-scale separate-effects mixing and natural circulation experiments to be performed at PSI. The areas of interest and investigation include the mixing of hot and cold liquids in open pools, the mixing and energy distribution within liquid pools resulting from the submerged injection (venting) of steam and gas mixtures, and the mixing of steam, nitrogen and, possibly, other gases in large, interconnected volumes.

In particular, this programme of work will support the PANDA experiments and provide additional help in scaling

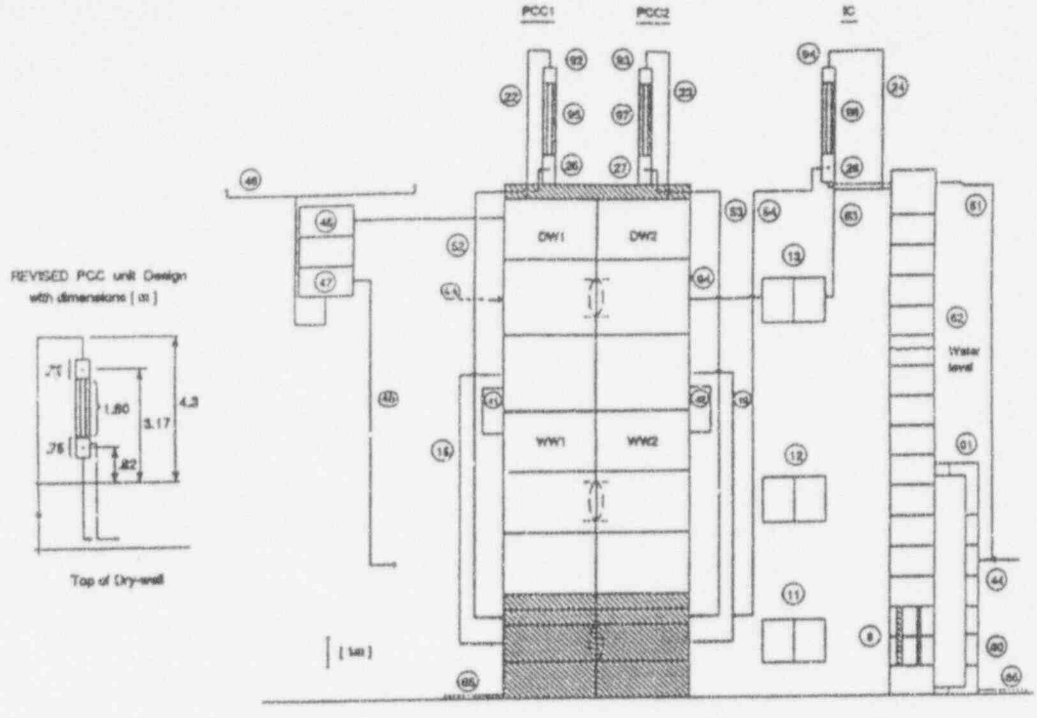


Figure 7: TRACG representation of PANDA.

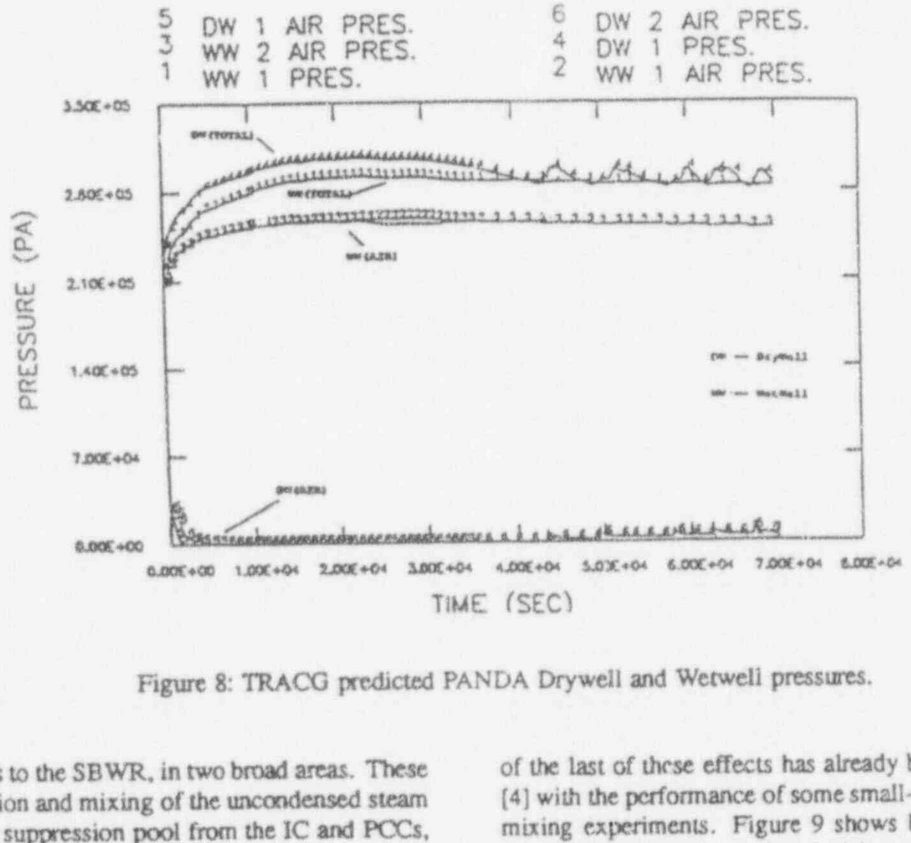


Figure 8: TRACG predicted PANDA Drywell and Wetwell pressures.

the PANDA results to the SBWR, in two broad areas. These are: the condensation and mixing of the uncondensed steam that flows into the suppression pool from the IC and PCCs, and the mixing of steam and nitrogen in the Drywell. In the first of the two areas described, there are several phenomena that will need to be investigated separately. For example, there is the condensation of the steam initially in the presence of the non-condensible gas nitrogen, and then there is the mixing of the resultant hot water with the bulk of the suppression pool as the hot water rises in a narrow buoyant plume to the pool surface. An initial investigation of the last of these effects has already been initiated at PSI [4] with the performance of some small-scale thermal plume mixing experiments. Figure 9 shows both a schematic of the plexiglas tank and electrical heater used in these experiments, and examples of the resultant rise in the water temperature as the water heated by the electrical heater rises in a very narrow plume to the pool surface and then spreads down in a 1-dimensional manner as the hot water replaces the cold water entrained in the rising plume. These experiments are being analysed using simple 1-dimensional models and 3-D CFD codes [4,5].

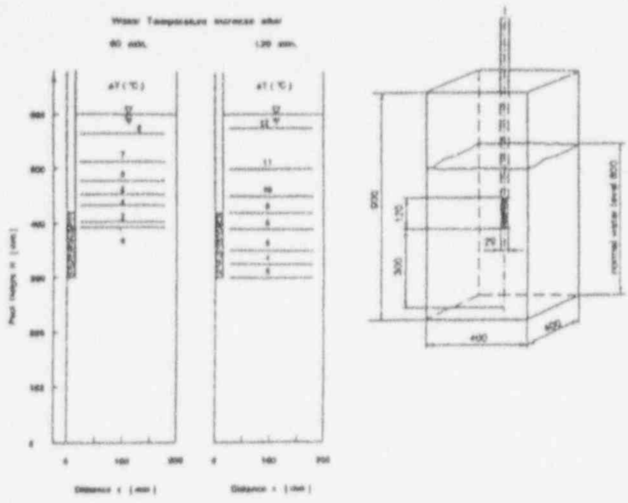

Figure 9: Mixing experiments; Facility and Results.

## 5 The AIDA Tests

Under severe accident conditions, fission products in the form of aerosols may escape from the RPV into the various compartments of the reactor containment. It is therefore possible that the IC and PCC units, which remove the decay heat, may be subjected to aerosols. The possible formation of an aerosol layer on the inside tube surface may affect the heat removal characteristics of the system, and plugging of some tubes may substantially degrade the condenser efficiency. The long-term pressurisation of the SBWR containment, following a postulated severe accident, depends on the continued function of the IC and PCC units, and this in turn on their aerosol behaviour. The goals of the AIDA programme are to:

(a) Experimentally determine the degree of IC and PCC condensation degradation in the presence of aerosols.

(b) Investigate aerosol behaviour under strong condensation in condenser tubes.

(c) Provide the basis for the development of a physical model for aerosol transport in the IC and PCC units.

An aerosol testing facility, including a generation system, is under construction. The plasma torches used for aerosol generation will produce up to three aerosol components (CsI, CsOH and MnO) with a maximum concentration of 10 mg/m$^3$, 0 to 100% steam to total gas (steam and noncondensibles) ratio, a gas flow rate of up to 9000 N l/min, and a system pressure of up to 5 bar.

A single tube, full height, glass model, subjected to realistic boundary conditions, will be constructed first, to visualise aerosol behaviour in a condensing environment. Two further test sections are planned to produce quantitative data for aerosol deposition and retention in the upper dome and tubes of the condenser units.

## 6 Summary and Conclusions

The major new experimental and analytical programme ALPHA initiated at the Paul Scherrer Institute has been briefly described. This programme is aimed at understanding long-term decay heat removal and aerosol questions for the next generation of Light Water Reactors. The ALPHA project includes four major items: the large-scale, integral system behaviour test facility PANDA; an investigation of the thermal hydraulics of natural convection and mixing in pools and large volumes (LINX); a separate-effects study of aerosol transport and deposition in plena and tubes (AIDA); while finally, data from the PANDA facility and supporting separate effects tests will be used to develop and qualify models and provide validation of relevant system codes.

PANDA will consist of a 1.5MW heat source and a number of large pressure vessels that can be interconnected by external piping to represent a variety of reactor containments. In the first instance, PANDA will be used to simulate the response of the SBWR containment to a Loss-Of-Coolant Accident. The SBWR uses two types of condensers (PCC and IC units) to remove the reactor decay heat to an external water tank, and PANDA will represent, on a 1/25 volumetric scale, the SBWR RPV, Drywell, Wetwell, and condensers. The PANDA facility has been designed to capture the asymmetric behaviour of the various IC and PCC units, arising from the non-uniform spatial distribution of non-condensible gases, and their influence on the condensation process. It therefore uses two large tanks to represent the SBWR annular geometry of the Drywell and Wetwell. The scaling factor for PANDA (1/25 volumetric) was determined on the basis that this is the minimum that provides an adequate simulation of the condenser pool-side behaviour. To aid the design of the PANDA facility, and to understand how it might respond to a simulated Loss-Of-Coolant Accident in the SBWR, the TRACG code is being used to model a variety of PANDA transients. The TRACG simulation of PANDA also provides a way of comparing the system behaviour of PANDA with that of the SBWR.

It is recognised that no scaled experiment can possibly provide a perfect simulation of all aspects of the physical behaviour of a full scale system. In response to this, and to the fact that two areas of particular importance in determining the SBWR containment pressure are the mixing of the nitrogen (and other non condensible gases) and the steam in the Drywell and the mixing of the uncondensed steam flowing into the Wetwell water pool, a companion, separate-effects program (LINX) was also initiated. LINX comprises both small- and large-scale experiments and analytical work, using simple 1-dimensional methods and 3-D CFD codes, to investigate natural circulation and mixing, of single- and multi-phase/multicomponent systems in large pools.

Under severe accident conditions, fission products in the form of aerosols may escape from the RPV into the various compartments of the reactor containment. It is possible that the IC and PCC units which remove the decay heat, may be subjected to aerosols. The possible formation of an aerosol layer on the inside tube surface may affect the heat removal characteristics of the system, and plugging of tubes may degrade the condenser efficiency. The AIDA program is being set up to investigate these phenomena using an aerosol generator feeding several small-scale test sections.

In conclusion, it is considered that the various elements of the ALPHA program will greatly enhance the understanding of the response of the SBWR containment and other similar concepts to Loss-Of-Coolant and other accidents, and will provide a large-scale experimental facility that can be used for similar studies of other reactor systems.

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# ATTACHMENT 7

# PANTHERS TEST

CONTAINS DRAFT PRESENTATION TO REASSESSMENT OF PF BILLIG

# PANTHERS TEST PROGRAM OVERVIEW

SBWR TECHNOLOGY REASSESSMENT

JUNE 21, 1994

# PANTHERS
# Performance Analysis and Testing of Heat Removal Systems

TOPICS:

- PROGRAM OBJECTIVES
- HEAT EXCHANGER DESIGN
- TEST FACILITY DESCRIPTION

# PROGRAM OBJECTIVES

- PROTOTYPE HEAT EXCHANGER PERFORMANCE DEMONSTRATION
  - THERMAL HYDRAULIC PERFORMANCE
  - LIFETIME STRUCTURAL INTEGRITY
- TRACG QUALIFICATION DATA
  - OVERALL HEAT EXCHANGER PERFORMANCE
  - LOCAL HEAT FLUX MEASUREMENTS
  - EFFECT OF NON-CONDENSABLES

# HEAT EXCHANGER DESIGN

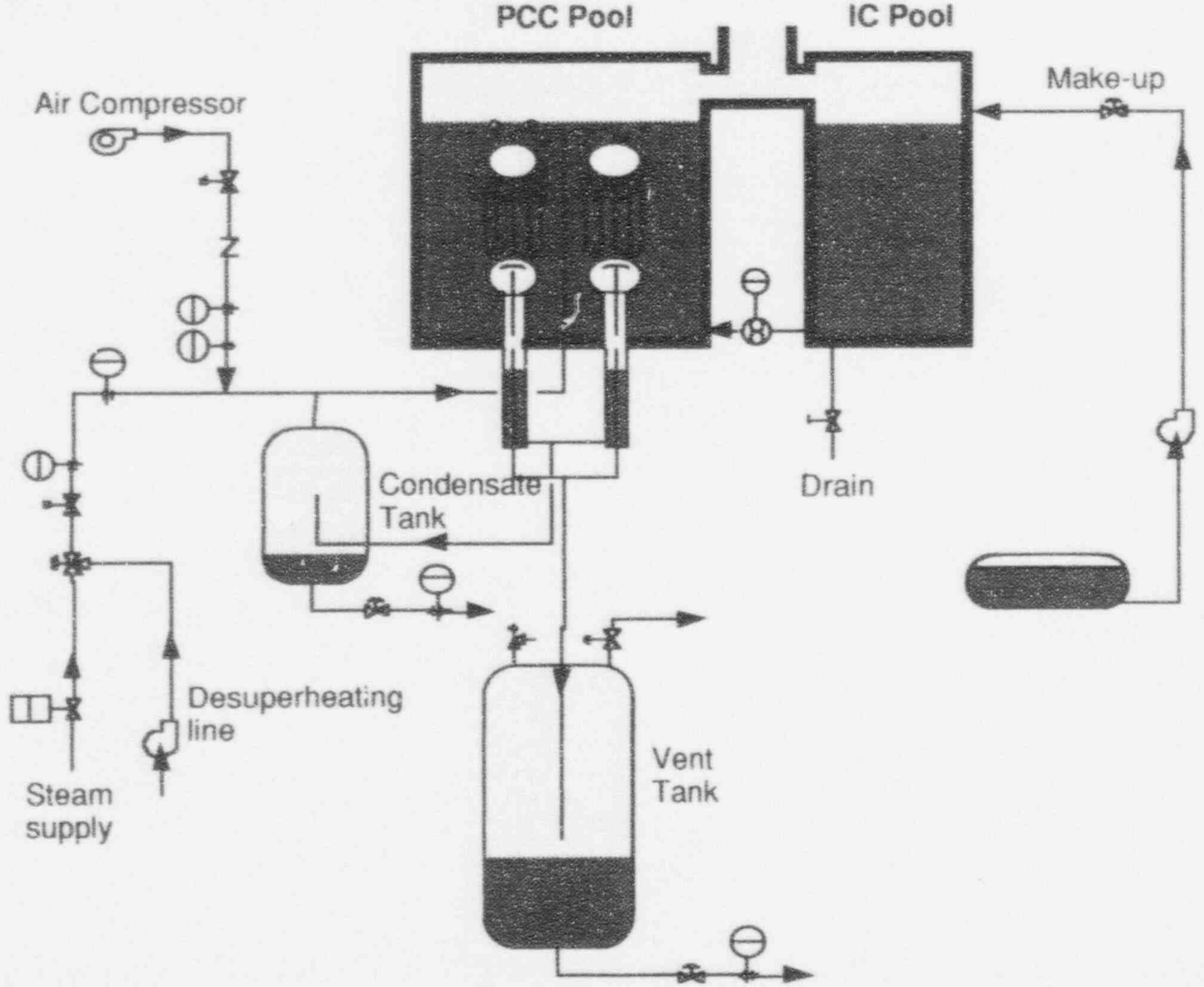
- DESIGNED AND MANUFACTURED BY ANSALDO
- VERTICAL TUBE TYPE
- PASSIVE CONTAINMENT COOLER (PCC)
  - DUAL MODULE (FULL UNIT)
  - FULL HEIGHT
  - STAINLESS STEEL CONSTRUCTION
  - PROTOTYPE MANUFACTURING PROCESSES
- ISOLATION CONDENSER (IC)
  - SINGLE MODULE (HALF UNIT)
  - FULL HEIGHT
  - INCONEL-600 CONSTRUCTION
  - PROTOTYPE MANUFACTURING PROCESSES

# TEST FACILITY DESCRIPTION

- PERFORMED AT SIET (PIACENZA, ITALY)

| PLANT CAPABILITIES | PCC | IC |
|---|---|---|
| MAX THERMAL POWER | 13 MW | 20 MW |
| MAX OPERATING PRESS | 0.9 MPa | 9.7 MPa |
| MAX OPERATING TEMP | 190 C | 320 C |
| MAX NON-CONDENSABLE FLOWRATE | 2 KG/SEC | 2KG/SEC |

# Schematic of PCC Test Facility



# PCC INSTRUMENTATION

- MASS FLOW RATE
  - FLOW MEASUREMENTS :
    - STEAM SUPPLY
    - NON-CONDENSABLE SUPPLY
    - POOL WATER SUPPLY
    - CONDENSATE TANK DISCHARGE
    - VENT TANK DISCHARGE
    - VENT TANK NON-CONDENSABLE DISCHARGE
- LIQUID LEVEL
  - LEVELS MEASURED ON LARGE TANKS
    - CONDENSATE TANK
    - PCC POOL
    - IC POOL
    - VENT TANK

# PCC INSTRUMENTATION (CONTINUED)

- FLUID TEMPERATURES
  - THERMOCOUPLES ON:
    - MAIN STEAM LINE
    - NON-CONDENSABLE SUPPLY LINE
    - INLET LINE DOWNSTREAM OF MIXING SECTION
    - INLET LINE UPSTREAM OF PCC POOL INLET SECTION
    - CONDENSATE TANK DISCHARGE LINE
    - PCC POOL - IC POOL CONNECTING LINE
    - PCC POOL (MANY LOCATIONS)
- PCC TUBE TEMPERATURES
  - 4 TUBES INSTRUMENTED:
    - INSIDE AND OUTSIDE WALL TEMPS
    - NINE AXIAL LOCATIONS

# PCC INSTRUMENTATION (CONTINUED)

- PRESSURE
  - PRESSURES MEASURED ON:
    - STEAM SUPPLY LINE
    - NON-CONDENSABLE SUPPLY LINE
    - INLET LINE DOWNSTREAM OF MIXING SECTION
    - INLET LINE UPSTREAM OF PCC POOL INLET SECTION
    - PCC POOL
- DIFFERENTIAL PRESSURE
  - DIFFERENTIAL PRESSURE MEASURED BETWEEN:
    - INLET LINE - UPPER HEADER
    - UPPER HEADER - LOWER HEADER
    - DRAIN LINE INLET - OUTLET
    - CONDENSATE TANK - UPPER HEADER
    - VENT LINE- VENT TANK
    - CONDENSATE TANK - VENT TANK

## PCC Instrumentation

SBWR

### Structural

| | Steam Pipe | Upper Header | Hx Tubes | Lower Header | Drain Pipe |
|---|---|---|---|---|---|
| Accelerometers | ✔ | ✔ | ✔ | | |
| LVDTs | ✔ | | | ✔ | |
| Scribe Marks | ✔ | | ✔ | | ✔ |
| Strain Gages | ✔ | ✔ | ✔ | ✔ | ✔ |
| Thermocouples | ✔ | ✔ | ✔ | ✔ | ✔ |

# PCC TEST CONDITIONS

- STEADY STATE PERFORMANCE TESTS
  - SATURATED STEAM
  - SATURATED STEAM /NON-CONDENSABLE GAS MIXTURES
  - SUPERHEATED STEAM
  - SUPERHEATED STEAM/NON-CONDENSABLE GAS MIXTURES
- PCC POOL EFFECTS TESTS
  - WATER LEVEL DECREASE
  - WATER ADDITION
- STRUCTURAL TESTS
  - 5 TIMES LIFE EXPECTED PRESSURE CYCLES
  - 10 LOCA CYCLES
  - 300 LEAK RATE TEST CYCLES

# PCC TEST PROCEDURES

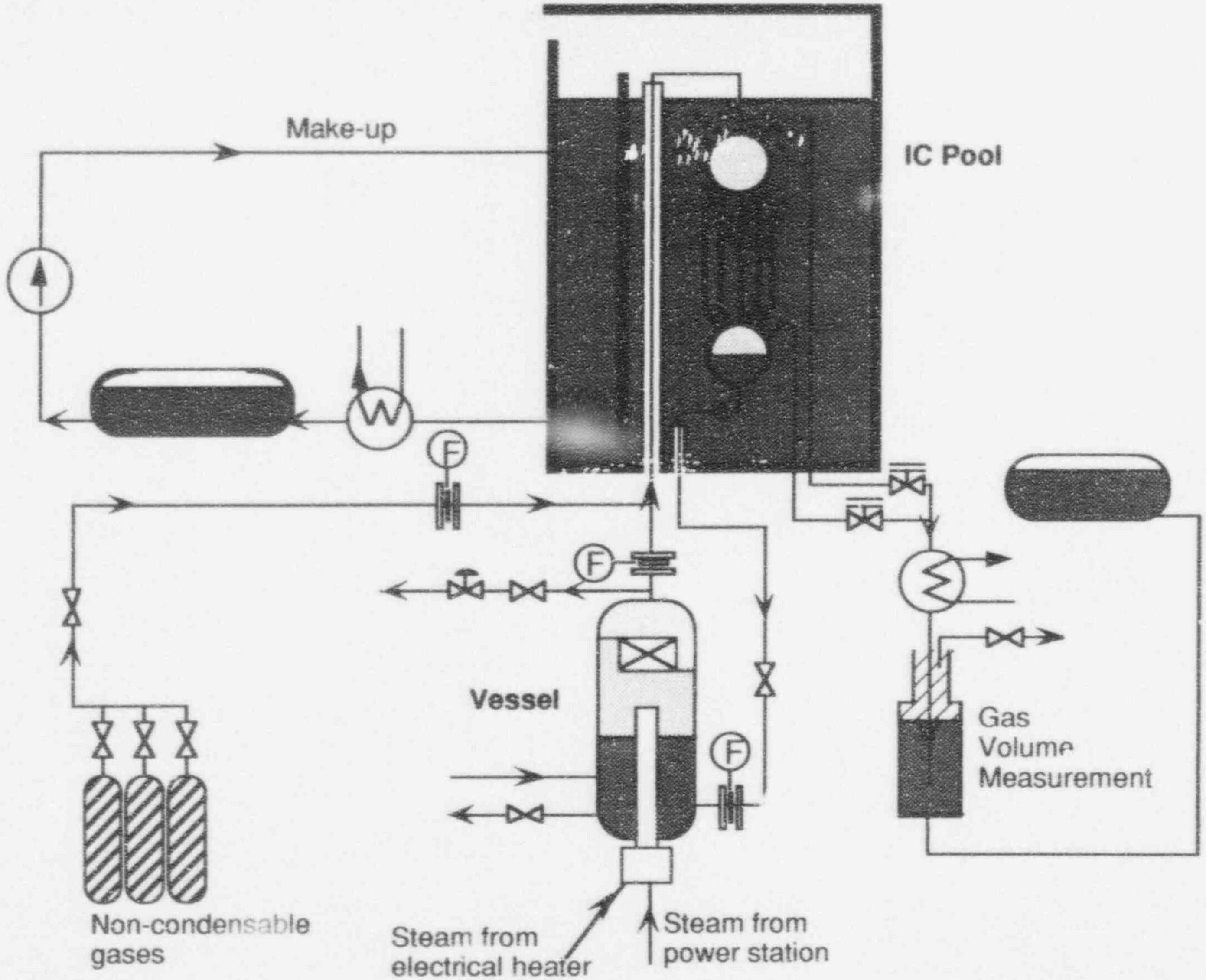
- STEADY STATE PERFORMANCE TESTS
  - SET INLET PRESSURE
  - SET INLET STEAM FLOW
  - SET INLET NON-CONDENSABLE FLOW
  - MEASURE HEAT EXCHANGER PERFORMANCE
- STRUCTURAL TESTS
  - LOCA TESTS  PRESSURIZED WITH STEAM
  - LEAK TEST CYCLES PRESSURIZED WITH AIR

# TEST DATA EVALUATION

- STEADY STATE PERFORMANCE TESTS
  - ASSURE HEAT EXCHANGER MEETS PERFORMANCE REQUIREMENTS
  - TRACG PREDICTIONS OF PERFORMANCE FOR SELECTED CONDITIONS
    - » DOUBLE BLIND PRE-TEST ANALYSIS
    - » POST TEST EVALUATIONS OF PREDICTIONS
- STRUCTURAL TESTS
  - LEAK TIGHTNESS FOLLOWING 5 TIMES ANTICIPATED 60 YEAR LIFE CYCLE

SBWR

# Schematic of IC Test Facility



# IC INSTRUMENTATION

- SIMILAR TO PCC
- EXCEPTION: NO TUBE WALL TEMPERATURE MEASUREMENTS

# IC TEST CONDITIONS

- FULL RANGE OF SBWR CONDITIONS TO BE SIMULATED
  - NORMAL IC OPERATION
  - NON-CONDENSABLE GAS EFFECTS
  - ATWS PERFORMANCE
  - FINAL TEST MATRIX IN PREPARATION

# IC TEST PROCEDURES

- STEADY STATE PERFORMANCE TESTS
  - SIMILAR TO PCC TESTS
- STRUCTURAL TESTS
  - PERFORM NDE BEFORE TESTING
  - SIMULATE ONE THIRD OF EXPECTED 60 YEAR PRESSURE/TEMPERATURE CYCLES
  - PERFORM POST TEST NDE

# IC TEST DATA EVALUATION

- STEADY STATE PERFORMANCE TESTS
  - ASSURE HEAT EXCHANGER MEETS PERFORMANCE REQUIREMENTS
  - TRACG PREDICTION OF PERFORMANCE FOR SELECTED TESTS
    - » DOUBLE BLIND PRE-TEST ANALYSIS
    - » POST TEST PREDICTION EVALUATION
- STRUCTURAL TESTS
  - NDE CONDITION EVALUATION

# ATTACHMENT 9

# SBWR SCALING

**CONTAINS REPORT NEDC 32288 BY G. YADIGAROGLU**

**NOTE: THIS REPORT CONTAINS ONLY THE TECHNICAL BASIS FOR SBWR SCALING. ADDITIONAL FACILITY UNIQUE INFORMATION WILL BE PRESENTED AT THE REVIEW**

NEDC-32288
Revision 0
Class 2
November 1993

# Scaling of the SBWR Related Tests

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# Nomenclature and Abbreviations

## Nomenclature

| | |
|---|---|
| $A$ | Surface area [$m^2$] |
| $a$ | Cross–sectional area [$m^2$] |
| $c_p$ | Specific heat at constant pressure [J/kg K] |
| $c_v$ | Specific heat at constant volume [J/kg K] |
| $D$ | Diameter [m] |
| $E$ | Internal energy [J] |
| $e$ | Specific internal energy [J/kg] |
| $f$ | Friction factor |
| $F$ | See Equation 2.27 |
| $F_n$ | See Equation 2.26 |
| $H$ | Height [m] |
| $h$ | Specific enthalpy [J/kg] |
| $h_{fg}$ | Latent heat of vaporization [J/kg] |
| $g$ | Acceleration of gravity [9.81 $m/s^2$] |
| $j$ | Volumetric flow rate [$m^3/s$] |
| $k$ | Ratio of specific heats, $c_p/c_v$ |
| $l$ | Length [m] |
| $L$ | Sum of lengths [m] |
| $M$ | Mass [kg] |
| $\dot{M}$ | Flow rate [kg/s] |
| $p$ | Pressure [Pa] |
| $\dot{Q}$ | Heat addition rate [W] |
| $R$ | Universal gas constant [J/kg K] |
| $R$ | System scale |
| $T$ | Temperature [K] |
| $t$ | Time [s] |
| $u$ | Velocity [m/s] |
| $V$ | Volume [$m^3$] |
| $v$ | Specific volume [$kg/m^3$] |
| $y$ | Mass fraction |
| $z$ | Axial coordinate [m] |
| $\delta$ | Kronecker delta |
| $\mu$ | Viscosity |

| | |
|---|---|
| $\Pi$ | Non-dimensional number |
| $\rho$ | Density [kg/m$^3$] |
| $\sigma$ | Surface tension |
| $\tau$ | Time constant [s] |
| $\omega$ | Characteristic frequency [s$^{-1}$] |

**Subscripts**

| | |
|---|---|
| G | Gas |
| L | Liquid |
| LG | Change liquid to gas |
| R | Scaling factor between prototype and model |

Additional subscripts are defined in the text

**Superscripts**

| | |
|---|---|
| + | Non-dimensional variable |
| o | Reference scale or variable |

# Abbreviations

| | |
|---|---|
| ADS | Automatic Depressurization System |
| BAF | Bottom of Active Fuel |
| DBA | Design-Basis Accident |
| BDLB | Bottom Drain Line Break |
| DPV | Depressurization Valve |
| DW | Drywell |
| h.t.c. | Heat Transfer Coefficient |
| GDCS | Gravity Driven Cooling System |
| GDLB | GDCS Line Break |
| H2TS | Hierarchical Two-Tier Scaling |
| IC | Isolation Condenser |
| ICS | Isolation Condenser System |
| LOCA | Loss-of-Coolant Accident |
| MIT | Massachusetts Institute of Technology |
| MSL | Main Steam Line |
| MSLB | Main Steam Line Break |
| NB | No-Break |
| NPP | Nuclear Power Plant |
| PCC | Passive Containment Cooler |

| | |
|---|---|
| PCCS | Passive Containment Cooling System |
| PIRT | Phenomena Identification and Ranking Table |
| PSI | Paul Scherrer Institute |
| RPV | Reactor Pressure Vessel |
| SBWR | Simplified Boiling Water Reactor |
| SC | Pressure Suppression Chamber |
| SP | Suppression Pool |
| SRV | Safety Relief Valve |
| TAF | Top of Active Fuel |
| UCB | University of California at Berkeley |

# ABSTRACT

This report 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 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 thermal plumes, mixing and stratification; heat and mass transfers 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 in the SBWR design. Finally, the scaling approach followed in designing the various SBWR-related facilities is reviewed in relation to the main purpose of the tests. The data collected from these facilities are used in the qualification of the system code TRACG.

# 1. Introduction

## 1.1 The SBWR and Related Tests

The SBWR uses gravity or natural circulation–driven, passive safety systems to provide emergency core coolant in case of a break in the primary system, to keep the core cooled and to remove decay heat from both the primary system and/or the containment. The main systems performing these tasks are the Gravity–Driven Cooling System (GDCS), the Isolation Condenser System (ICS), and the Passive Containment Cooling System (PCCS) (Vierow et al., 1992), Figure 1–1.

Emergency core cooling water is provided to the core by the GDCS. This system consists of three water pools situated above the top of the core, from which makeup coolant can flow by gravity to replenish the coolant lost from the Reactor Pressure Vessel (RPV). However, the GDCS can operate only after depressurization of the RPV; therefore, the SBWR is equipped with an Automatic Depressurization System (ADS) that performs this function. The depressurization of BWR primary systems is well understood, since it has been studied extensively in relation to the classical BWR designs. Indeed, the phenomena taking place during the early phase of blowdown inside the RPV have been extensively investigated by several series of tests; these constitute the basis for the corresponding qualification of the TRACG code (Andersen et al., 1993b). The containment loads during early blowdown have also been extensively investigated (GE, 1980; NRC, 1984; GE, 1987). The GDCS is, however, a relatively novel concept and requires some attention. The General Electric Company (GE) has therefore conducted the GDCS Integrated Systems Test (GIST) series of tests to investigate the behavior of the SBWR during the latter part of the depressurization phase. Proof of the technical feasibility of the GDCS concept was a major test objective.

Decay heat removal from the primary system while it is intact or under high pressure is performed by the ICS. The ICS consists of three Isolation Condensers (IC) located in a pool on top of the reactor building. When redundant squib valves are opened, steam from the primary system flows into the tubes of the ICs, condenses, and returns to the RPV, removing stored energy and is well understood, since such units have been in operation for many years in older BWRs. Thus, there is no specific need to experimentally verify the high–pressure operation of the SBWR decay heat removal system.

Decay heat is removed from the drywell (DW) by the PCCS, which employs three PCC condensers also located in interconnected IC pool compartments on top of the reactor building. The PCC condenser tubes are permanently connected to the DW. A

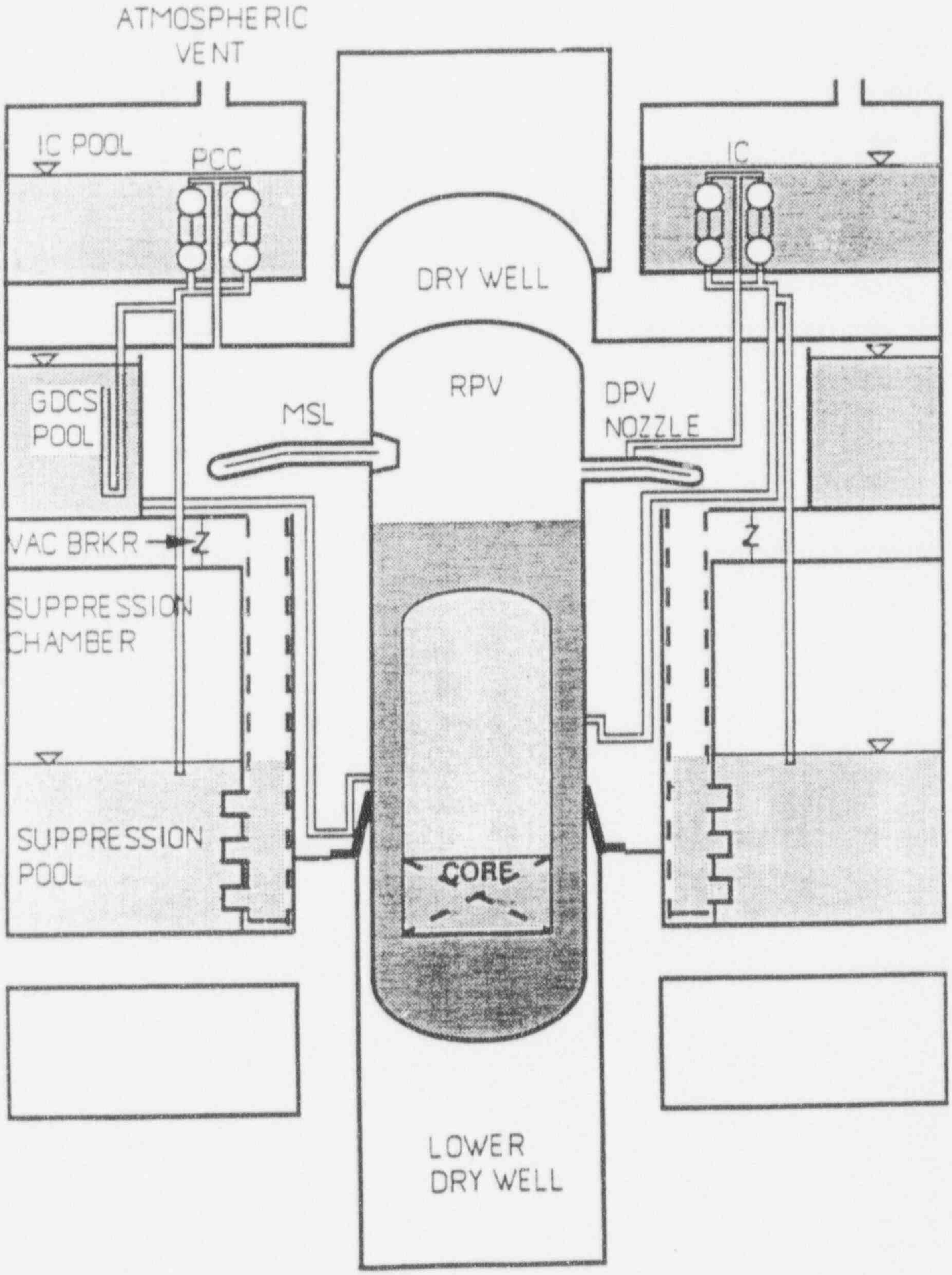


Figure 1-1 Passive Core and Containment Cooling Systems of the SBWR

mixture of steam and noncondensable gases (nitrogen present in the containment during normal operation) may enter the PCC condensers. The steam will condense, while the noncondensable gases must be vented to assure proper operation of the condensers. This is accomplished by conveying and venting the noncondensable gases into the suppression pool (SP) in the Suppression Chamber (SC) (or Wetwell).

Since the DW volume is connected directly to the SP either via the main pressure suppression vents or through the PCC condensers and their vent lines, the path that the steam will follow depends on the pressure differences between the DW volume and the two possible venting points. During the long-term containment cooling period, direct opening of the main vents and condensation of the steam in the SP must be avoided, since the SP is not provided with a safety-grade cooling system; the steam must be condensed in the PCC (or IC) condensers and any noncondensables vented to the SC. Although the operation of the condensers is understood, experimental verification of their integral, system behavior under a variety of conditions was deemed necessary. Two experimental facilities were provided for this purpose. The GIRAFFE facility, operated by Toshiba in Japan, provided extensive information about system behavior; this information was used to qualify the TRACG Code (Andersen et al., 1993a; Andersen et al., 1993b; Kim et al., 1993) for calculation of long-term decay heat removal and constitutes one of the major bases for certification of the SBWR (Vierow, 1993). The larger-scale PANDA facility, near completion at the Paul Scherrer Institute (PSI) in Switzerland, will provide additional information and will address issues such as the effects of the operation of several condenser units in parallel, the distribution of the constituents (steam and noncondensables) in the large DW volume, and mixing in the containment compartments. Availability of data from integral facilities having different scales is clearly an advantage for understanding system behavior and performing code qualification (Boucher et al, 1992). The PANDA experiments are part of the ALPHA program (Advanced LWR Passive Heat Removal and Aerosol Program) conducted at PSI.

The condensation of mixtures of steam and noncondensable gases in tubes under conditions expected in the PCC units has been investigated in experimental programs conducted at the Massachusetts Institute of Technology (MIT) (Siddique, 1992; Siddique et al., 1989, 1993) and at the University of California-Berkeley (UCB) (Vierow, 1990; Ogg, 1991; Vierow and Schrock, 1991). Full-scale tests of the IC and PCC units are being conducted at the PANTHERS facility at the SIET laboratory in Italy (Masoni et al., 1993).

Additional references about details of the various test facilities can be found in the letter by Marriott (1993). The design of all these experimental facilities and the conduct of the various tests was guided by consideration of the proper modeling and simulation of the various phenomena taking place. The objectives of this report are to:

(1) summarize the philosophy used in defining the SBWR-related experimental program;

(2) describe the rationale used in scaling the various SBWR subsystems in the experimental facilities;

(3) verify the scaling criteria and laws used for the various facilities; and

(4) provide assurance that the phenomena of importance in relation to GDCS performance and long-term decay heat removal from the containment were properly addressed or represented in the tests.

## 1.2 General Approach for Code Qualification, Testing and Scaling

The approach adopted is similar to the one used for most LWR safety-related large-scale integral tests. It is clear that system tests (such as the GIST, GIRAFFE and PANDA tests) do *not* have to provide exact system simulations of the prototype. In fact, it is neither practical nor desirable to attempt to provide such exact simulations. However, system tests do provide *data* covering all essential phenomena and system behavior under a variety of conditions, which are used to qualify a system code (in the particular case, the TRACG Code used for safety analysis by GE).

To obtain data in the proper range of system conditions, the relative importance of the phenomena and processes present in the tests should not differ significantly from what is expected to take place in the SBWR. Similarly, the overall behavior of the test facility should not diverge significantly from that of the SBWR; in particular, one should not observe bifurcations in system behavior leading to quite different intermediate or end states. Finally, the tests should provide sufficiently detailed information, obtained under well-controlled conditions, to provide an adequate and sufficient database for qualifying the system code, TRACG.

Following current practice (INEL, 1989), a Phenomena Identification and Ranking Table (PIRT) was prepared for the SBWR post-LOCA containment phenomena. A PIRT identifies the phenomena and processes that are of particular importance during the various phases of a postulated accident or class of accidents. These phenomena receive, then, particular attention during code qualification. The SBWR PIRT was used to identify the phenomena of importance in relation to scaling of the experimental facilities. These phenomena are listed in Section 3 of this report (Table 3-1), where the scaling of specific phenomena is addressed.

## 1.3 Scope and Objectives of the Scaling Study

The scope of the scaling study reported here was to:

- Describe the scaling philosophy and strategy used in designing the various tests.
- Provide the applicable scaling laws.
- Show that the test facilities properly "scale" the important phenomena and processes identified in the SBWR PIRT and/or provide assurance that the experimental observations from the test programs are representative of SBWR behavior.
- Identify any distortions in phenomenology and/or scaling and discuss their importance; in particular, identify the ways by which such omissions and/or scaling distortions can be considered when the experimental data are used for code qualification.
- Verify the applicability of the condensation heat transfer data obtained in the single–tube university tests for the SBWR safety analysis.
- Provide the basis for showing that the experimental data are sufficient for qualifying TRACG.

### 1.3.1 Accidents and Accident Phases Considered

The range of accidents considered includes the main steam line break, as well as other breaks of the primary system, such as the GDCS line break and the bottom drain line break.

The scenario of these accidents can be rougly subdivided into three phases:

- The *blowdown phase* extending from the initiation of rapid depressurization by blowdown up to the time of refill of the LOCA vents. The blowdown phase can be further subdivided into an *early* phase extending until the time the pressure reaches a level of about 0.8 MPa, and a *late* blowdown phase thereafter.
- An intermediate *GDCS phase* during which the GDCS is delivering its stored water inventory to the primary system[a].
- A *long–term cooling phase* beginning when the GDCS inventory starts becoming replenished by the condensate flowing down from the ICS and PCCS (i.e., when the GDCS hydrostatic head necessary to drive flow into the core is made up by the ICS and PCCS condensate). At about the same time, the ICS and PCCS condensers become the dominant decay heat removal mechanism, replacing the heat sink provided by the water inventory initially stored in the GDCS pools.

The scaling analysis performed in this report is directed mainly at the scaling of reactor and containment components and phenomena which are significant over the time period starting with the latter blowdown phase and extending into the long-term cooling phase (see Section 1.3.1). As stated in the Introduction (Section 1.1), phenomena associated with the early stage of depressurization of a BWR vessel are well understood and are not considered to be part of the SBWR testing program. Thus, this report deals with *post-LOCA containment performance*.

### 1.3.2 Important Safety Issues

The tests conducted in relation to the SBWR are aimed at answering certain safety related issues, including:

- The possibility of core uncovery and damage — this issue is clearly related to the water inventory in the RPV resulting from the flows out of the break and from the GDCS, and to the RPV depressurization rate. This issue was addressed with the GIST tests, which demonstrated the feasibility of depressurizing the reactor core to sufficiently low pressures to enable reflooding by the gravity-driven flow from an elevated pool (Billig, 1989).
- Limitation of the containment pressure. This issue is related to the capability of the ICS and PCCS to remove decay heat. The distribution of phases in the various containment compartments and the temperature at the surface of the SP are significant variables.
- Effectiveness of condenation of steam injected into the SP from the PCCS vents.
- The performance of certain key containment components, such as: (1) the cyclic performance of the PCCS (in relation to venting of noncondensables), as observed in the GIRAFFE tests; (2) the intermittent opening of the vacuum breakers; and (3) the possible opening of the main vents during long-term containment cooling.
- The heat transfer performance of the ICS and PCCS condensers — this depends on both condensation heat transfer inside the condenser tubes in the presence of noncondensables and on heat transfer at the outside surface of the tubes in the pool, including IC pool inventory, temperature, and circulation rate. The possible degradation of the performance of the PCCS condensers due to insufficient venting and the accumulation of noncondensables in their tubes is also of importance.
- Structural integrity of the ICS and PCCS condenser units.

The issues identified above are addressed by the GIST, GIRAFFE, PANDA, and PANTHERS tests.

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[a] For certain scenaria, GDCS flow may start before the end of the blowdown phase.

## 1.4 The H2TS Scaling Methodology

The NRC, in relation to Severe Accident Research, has developed a "structured" scaling methodology, which "provides the confidence that scaled experiments faithfully reproduce the phenomena which will occur in a NPP" (NRC, 1991). This methodology, referred to as Hierarchical, Two–Tiered Scaling (H2TS), addresses the scaling issues in two tiers: a top–down (inductive) system approach, followed by a bottom–up (deductive) process–and–phenomena approach. The method uses characteristic time ratios and a hierarchical characterization of the processes according to their temporal and spatial scales. To establish a hierarchical architecture for the system considered, one proceeds with a physical decomposition according to interacting subsystems, modules, constituents (materials), phases, and their geometrical configuration.

The H2TS methodology is applied to the extent necessary and practical in this work. Indeed, several scaling considerations that are addressed by H2TS are automatically satisfied in the SBWR–related experiments, in which the fluids and their thermodynamic states are prototypical (water and noncondensable gases under the pressures, temperatures, and concentrations expected in the SBWR). Moreover, the experiments are conducted at scales at which "microscopic" level interactions between phases (e.g., *local* mixing of gases) are not expected to be affected by the scale of the experiments. Thus, several hierarchical levels, related to constituents and phases, need not to be addressed.

## 1.5 Scaling Issues for the SBWR Related Tests

The experimental program supporting SBWR safety analysis includes the tests listed in Table 1–1, together with their volume scales in relation to the actual SBWR.

All these tests were (or will be) conducted under the following conditions:

- Actual fluids (water and steam, noncondensables, with the exception of substitution of air for nitrogen in most tests and of helium for hydrogen in all tests where hydrogen presence was simulated).
- Prototypical initial thermodynamic state of the fluids or mixtures (pressure, temperature, component concentrations).
- Full height.
- Test facilities are large enough (i.e., pipe and vessel dimensions have a sufficiently large characteristic length scale) so that "microscopic" level

**Table 1-1**

**The SBWR Related Tests**

| Test | Purpose | Volume Scale |
|---|---|---|
| GIST | Integral GDCS system test | 1/508 |
| GIRAFFE<br>PANDA | Integral long-term containment heat removal tests | 1/400<br>1/25 |
| PANTHERS | Structural and heat transfer tests of the ICS and PCCS condensers | Full-scale prototypes |
| UCB<br>MIT | Condensation in the presence of noncondensables | Single-tube (near full-scale) |

interactions between the phases (e.g., *local* mixing of two different gases) are not expected to be scale dependent.

The geometrical "macroscopic" level configuration of the phases needs to be considered, however, and leads to the requirement of preservation of the large-scale mixing behavior of fluids in single-phase situations, and of flow regimes in two-phase flow situations. The large-scale mixing issues are addressed in subsequent sections of this report. Scaling requirements to preserve flow regimes are discussed by Schwartzbeck and Kocamustafaogullari (1988). For the SBWR-related tests considered here, the geometrical scale of the models was sufficiently large so that important flow regime distortions are not expected; in addition, most containment flows are single-phase.

These considerations and design requirements remove any hierarchical concerns regarding constituents and phases, as mentioned above. Moreover, the full-height design of the experimental facilities leads to proper simulation of the natural gravity heads that are essential for the natural circulation systems and loops considered here. The remaining *geometrical* scaling issues are addressed in this report.

Additional scaling issues examined in this report include: (1) scaling of phenomena and processes; (2) multidimensionality, and (3) multi-unit, multi-element operation.

### 1.5.1 Scaling of Phenomena and Processes

The influence of spatial scale on phenomena and processes is considered in a bottom-up fashion for those ranked as important in the SBWR PIRT (e.g., stratification in pools and the development of thermal plumes).

### 1.5.2 Multidimensionality - Non Uniform Distribution Effects

This issue is related to the large differences in spatial scales between experiments like GIRAFFE and GIST and the SBWR. One must ensure that non-homogeneities in the distribution of constituents or phases that may be occurring at a particular scale are understood (and/or "scaled" properly whenever possible). The issue is addressed (Section 3.2) by running counterpart tests in facilities having different scales (GIRAFFE and PANDA) and by examining the physical reasons that may lead to such non-homogeneities in a phenomenological, bottom-up fashion.

### 1.5.3 Multi-Unit, Multi-Element Operation

The SBWR has multiple key components such as the ICS and PCCS condensers and vent lines. Moreover, the condensers have a large number of similar elements (tubes). The exact numbers of units, or elements per unit, cannot be duplicated in the experiments, and this raises the question of possible dissimilar, non-symmetric operation of the units or elements and its effects on system performance. Again the issue is addressed by analysis and by running tests in facilities having a range of number of tubes or units in parallel: (1) single-tube university tests; (2) three-tube GIRAFFE tests; (3) 20-tube, 4-unit PANDA tests; and (4) testing of entire full-scale modules in PANTHERS.

## 1.6 System Considered

### 1.6.1 Subdivision of the System into Subsystems and Components

For the purposes of this study, the SBWR System is subdivided into the subsystems or components shown on Table 1-2; their scaling by class of subsystem is considered in this report. Interactions (in this particular case, essentially transfers of mass and energy) between components are also a scaling consideration. The remaining SBWR systems or components are not relevant to this study and thus are not considered here.

**Table 1-2**

**SBWR Subsystems and Components Considered**

Reactor Pressure Vessel, RPV
Main Steam Lines (MSL) and Depressurization Valves (DPV)

Drywell, DW
- Upper DW volume
- DW annular volume surrounding RPV
- Lower DW volume below RPV skirt

Suppression Chamber, SC
- Gas space
- Liquid volume in suppression pool (SP)

Main (LOCA) vents connecting DW to SP (8)
Vacuum breakers between DW and SC (3)
Leakage path between DW and SC

Gravity-Driven Cooling System (GDCS) pools (3)
- Gas space
- Liquid space

Equalization line with check valve connecting SP to RPV (3)

Isolation Condenser System (ICS) condensers (3)
Passive Containment Cooling System (PCCS) condensers (3)
Noncondensable PCCS vent lines from condensers to SP (3)
Isolation Condenser Pool with interconnected subcompartments

Other lines connecting the various subsystems listed above.

### 1.6.2 Fluids and Other Materials

The *differences* between prototypical fluids and other materials that enter into consideration are:

- Air is used instead of nitrogen in the PANTHERS, GIST, and PANDA system tests and in the UCB and MIT single-tube tests.
- Helium is used to simulate hydrogen in all related tests.
- The wall materials used in the SBWR and in the various integral facilities are different. This issue is discussed in Section 3.4, which deals with the heat capacity and conduction in containment structures.

# 2. General Scaling Considerations – Top–Down Approach

The SBWR and the corresponding scaled test facilities are referred to generically and collectively as the "System" or "SBWR System" in this report. Alternatively, the SBWR and a particular test facility are referred to, following common practice in scaling studies, as the "prototype" and the "model," respectively.

The *general* scaling criteria applicable to the SBWR System with its various subsystems and components and their counterparts in the related tests under consideration are derived in this section by a top–down approach. General scaling criteria have been derived by several authors (e.g. Ishii and Kataoka, 1983; Kocamustafaogullari and Ishii, 1984; Kiang, 1985; Boucher et al. 1991). Generally, these are not specific to the combined thermodynamic and thermal–hydraulic phenomena taking place inside containments and therefore are not directly applicable here. To arrive at general scaling criteria applicable to the SBWR System, the controlling processes in generic subsystems having the essential characteristics of *classes* of SBWR subsystems (e.g. containment volumes, pipes, etc.) are considered.

The SBWR System consists of a number of volumes (RPV, DW, SC, etc.) connected via junctions (i.e., openings, piping, vents, heat exchanging equipment — e.g., the ICS and PCCS condensers — fans, etc.). Mass and energy transfers take place between these volumes through their junctions. Heat may also be exchanged between volumes by conduction through the structures connecting them. These exchanges lead to changes in the thermodynamic condition of the various volumes; this, in particular leads to changes of the volume pressures. The junction flows (flows between volumes) are driven by the pressure differences *between* volumes. Thus, the *thermodynamic behavior* of the system (essentially, its *pressure history*) is linked to its *thermal–hydraulic behavior* (the flows of mass and energy between volumes). Proper scaling of these processes is the goal of the SBWR–related tests considered here and the topic addressed in this section.

The generic processes considered in this section are:

(1) The effects of the addition of heat and mass to a gas or liquid volume (namely, the resulting rates of change of the pressure).

(2) The rates of phase change at interfaces such as pool surfaces.

(3) Flows of mass between volumes.

Prototypical fluids under prototypical thermodynamic conditions were used in all the SBWR–related tests. The fact that the fluids are expected (by design and operation of the test facilities) to be in identical states in the prototype and the models, will be used to simplify the following analyses.

For the top–down approach taken in this section, it is further assumed that the *composition* of the various fluids in different parts of the models (e.g., the fraction of noncondensables) remains also prototypical. The conditions that must be satisfied to ensure this additional requirement are examined in the following bottom–up study (Section 3), in which the particular phenomena affecting the *distribution* of the fluid composition are considered. One such example is stratification that may affect the composition of the fluids being transferred between volumes or the conditions at interfaces between liquid and gas volumes.

The first two processes listed above (1 and 2) confirm, as shown below, the validity of the (familiar) scaling of all the following variables with the "system scale" :

power : volume : horizontal areas in volumes : mass flow rate

A time scale of 1:1 between prototype and models has been adopted for all tests; however, this is not a necessity. Under certain conditions, the choice of a scale for the volumes different from the "system scale" will lead to accelerated (or decelerated) tests in time; this possibility is discussed in Section 2.4.4.

Process (3) will lead to the determination of the pressure drops and of the hydraulic characteristics of the junctions between volumes. In the SBWR System, certain pressure drops and the corresponding junction flows are controlled by the submergence depth of vents. The analyses of these processes will justify the choice of 1:1 scaling for the vertical heights in general and for the submergence depths in particular.

The pressure evolution resulting from the thermodynamics of the system and the pressure drops between volumes must clearly be scaled in an identical fashion. Considering the fact that prototypical fluids are used, this requirement links the properties of the fluid (in particular the latent heat and the specific volumes of water and steam) to the pressure differences between volumes (and to the submergence depths of vents), resulting in 1:1 scaling for pressure drops. Thus, the above considerations result in:

1:1 scaling for pressure differences, elevations, and submergences

This scaling rule determines the pipe diameters, lengths, and hydraulic resistances, and also determines the transit times between volumes. These transit times should, in principle, have the same (1:1) time scale as the inherent time constants of the system considered in the analysis of process (1). This matching cannot be perfect, but it is shown (Section 2.4.1) not to be important.

## 2.1 Thermodynamic Evolution of Containment Volumes with Mass and Energy Additions

Consider the control volume V of Figure 2–1 containing a mass M with internal energy E at a pressure p and a temperature T. The volume contains a number of constituents (noncondensable gases, steam, etc.) each denoted by the subscript j. Any changes in the kinetic and potential energy of the mass M are much smaller than changes in its intrinsic internal energy and therefore are neglected. The system is well mixed (i.e., the distributions of constituents and of the temperature are uniform, and at thermodynamic equilibrium).

The mass continuity equation for this volume is:

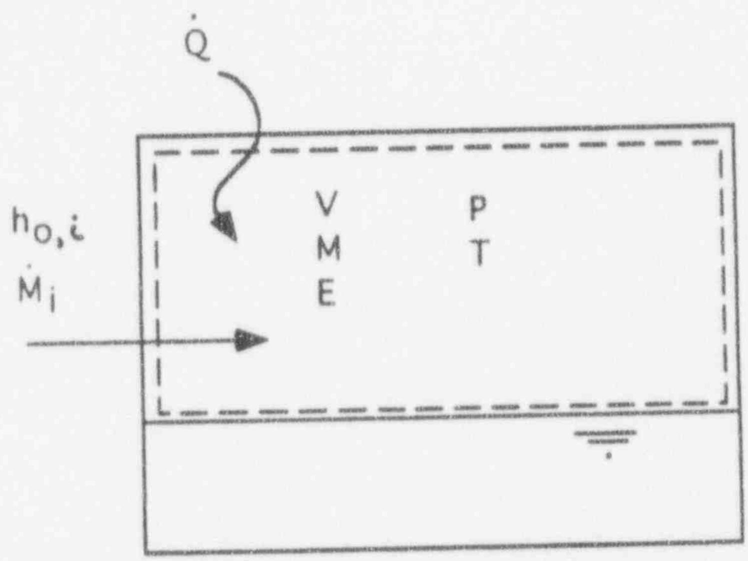



Figure 2–1 A Containment Volume (dashed line, in the case of a gas space) Receiving Mass Flow Rates $\dot{M}_i$ with Corresponding Total Enthalpies $h_{o,i}$, and Heat at the Rate $\dot{Q}$

$$\frac{dM}{dt} - \sum \dot{M}_i = 0 \tag{2.1}$$

where $\dot{M}_i$ are the mass flow rates entering the volume and carrying with them total enthalpy $h_{o,i}$. The total enthalpy (subscript o) includes the kinetic and potential energy of the various streams. The energy conservation equation is:

$$\frac{dE}{dt} = - p\frac{dV}{dt} + \dot{Q} + \sum \dot{M}_i h_{o,i} \tag{2.2}$$

where $\dot{Q}$ is the heat added to the system (e.g., by conduction through the wall). *Phase changes* taking place at interfaces bounding the control volume are considered, since these bring mass flow rates and enthalpies included in the $\sum \dot{M}_i$ and $\sum \dot{M}_i h_{o,i}$ terms.

The purpose here is to derive the equation relating the rate of change of the volume pressure dp/dt to the mass, enthalpy and heat additions (see, e.g., Section 2.14 of Moody, 1990). The specific internal energy of the system,

$$e = \frac{E}{M}$$

is a function of two thermodynamic variables, chosen here to be the pressure and the specific volume $v = V/M$, and of the mass fractions $y_j$ of the various constituents:

$$e = e(p, v, y_j)$$

with

$$\sum y_j = 1$$

The differential of e can be calculated as:

$$de = \left.\frac{\partial e}{\partial p}\right|_{v,y_j} dp + \left.\frac{\partial e}{\partial v}\right|_{p,y_j} dv + \sum \left.\frac{\partial e}{\partial y_j}\right|_{p,v,y} dy_j \tag{2.3}$$

where the subscript $y_j$ means that all $y_j$ are kept constant, while the subscript y denotes that all $y_j$ except the one appearing in the partial derivative are kept fixed. We note also that

$$E = Me \quad \text{and} \quad V = Mv$$

and therefore

$$\frac{dE}{dt} = \frac{d}{dt}(Me) = M\frac{de}{dt} + e\frac{dM}{dt} \tag{2.4}$$

$$\frac{dV}{dt} = M\frac{dv}{dt} + v\frac{dM}{dt} \tag{2.5}$$

Combining Equations 2.1 through 2.5:

$$\frac{dE}{dt} = M \left.\frac{\partial e}{\partial p}\right|_{v,y_j} \frac{dp}{dt} + M \left.\frac{\partial e}{\partial v}\right|_{p,y_j} \frac{dv}{dt} + M \sum \left.\frac{\partial e}{\partial y_j}\right|_{p,v,y} \frac{dy_j}{dt} + e\sum \dot{M}_i$$

$$= - pM\frac{dv}{dt} - pv\sum \dot{M}_i + \dot{Q} + \sum \dot{M}_i h_{o,i} \tag{2.6}$$

Solving this equation for dp/dt and using again Equations 2.5 and 2.1:

$$\frac{dp}{dt} = \frac{\sum\left[\dot{M}_i\left(h_{o,i} - e + v \left.\frac{\partial e}{\partial v}\right|_{p,y_j}\right)\right] + \dot{Q} - \left[p + \left.\frac{\partial e}{\partial v}\right|_{p,y_j}\right] \frac{dV}{dt} - \frac{V}{v} \sum\left[\left.\frac{\partial e}{\partial y_j}\right|_{p,v,y} \frac{dy_j}{dt}\right]}{\frac{V}{v} \left.\frac{\partial e}{\partial p}\right|_{v,y_j}} \tag{2.7}$$

We note that this equation yields the rate of change of the pressure in terms of the heat addition, of the mass and enthalpy fluxes into the volume, and the changes of volume composition. The rate of change of the volume dV/dt (e.g., due phase change) is also considered. The partial derivatives of e with respect to v, p and the mass fractions $y_j$, as well as their combinations with other thermodynamic variables, are *thermodynamic properties* of the particular mixture contained in the volume. Thus, the following short–hand notations for certain quantities appearing in Equation 2.7,

$$e^* \equiv e - v \left.\frac{\partial e}{\partial v}\right|_{p,y_j}$$

$$p^* \equiv p + \left.\frac{\partial e}{\partial v}\right|_{p,y_j}$$

$$f_{1,j} \equiv \frac{1}{v} \left.\frac{\partial e}{\partial y_j}\right|_{p,v,y} \quad \text{(units of energy per unit volume)}$$

$$f_2 \equiv \frac{1}{v} \left.\frac{\partial e}{\partial p}\right|_{v,y_j} \qquad \text{(non dimensional)} \tag{2.8}$$

denote thermodynamic properties of the mixture, which are functions of p, v, and the $y_j$. When prototypical fluids under prototypical thermodynamic conditions are used, these thermodynamic properties are identical for prototype and model.

The mass flow rates can be expressed as products of the volumetric flow rates $\dot{J}$ times the corresponding densities:

$$\dot{M}_i = \dot{J}_i \ \rho_i$$

Thus, Equation 2.7 takes the simpler form:

$$\frac{dp}{dt} = \frac{\sum \left[\dot{J}_i \rho_i (h_{o,i} - e^*)\right] + \dot{Q} - p^* \frac{dV}{dt} - V \sum \left[f_{1,j} \frac{dy_j}{dt}\right]}{V \ f_2} \tag{2.9}$$

The following reference quantities (denoted by the superscript $^o$) are used:

- For volume: $V^o$
- For volumetric flow rates: $\dot{J}^o$
- For heat addition: $\dot{Q}^o$
- For densities: $\rho^o$
- For pressure, a reference pressure difference: $\Delta p^o$
- For enthalpies and internal energies, a reference enthalpy difference: $\Delta h^o$

A reference time is obtained by combining the volume and volumetric flow rate scales:

$$\tau^o = \frac{V^o}{\dot{J}^o} \tag{2.10}$$

This reference time $\tau^o$ is the *volume fill time* (or *residence time*) for mass flowing into the volume $V^o$ at the volumetric flow rate $\dot{J}^o$ (Zuber, 1991).

Equation 2.9 will be non–dimensionalized by dividing the dimensional variables z by the reference values $z^o$ above; this produces the non–dimensional variables $z^+$:

$$z^+ \equiv \frac{z}{z^o} \tag{2.11}$$

In particular, note that

$$e^* = e^{*+} \Delta h^o$$

$$p^* = p^{*+} \Delta p^o$$

$$f_{1,j} = f_{1,j}^+ \Delta h^o \rho^o \tag{2.12}$$

The mass fractions $y_j$ and $f_2$, being non–dimensional, require no scaling:

$$y_j = y_j^+ \qquad \text{and} \qquad f_2 = f_2^+$$

By non–dimensionalization, Equation 2.9 takes the form:

$$\frac{dp^+}{dt^+} = \frac{1}{f_2^+ V^+} \left[ \Pi_{hp} \sum \left[ \dot{j}_i^+ \rho_i^+ (h_{o,i}^+ - e^{*+}) \right] + \Pi_1 \dot{Q}^+ - p^{*+} \frac{dV^+}{dt^+} - \Pi_{hp} \sum \left[ f_{1,j}^+ \frac{dy_j^+}{dt^+} \right] \right] \tag{2.13}$$

Two non–dimensional groups appear in the equation above:

$$\Pi_1 \equiv \frac{\dot{Q}^o \tau^o}{\Delta p^o V^o} \tag{2.14}$$

and

$$\Pi_{hp} \equiv \frac{\Delta h^o}{\Delta p^o / \rho^o} \tag{2.15}$$

$\Pi_{hp}$ can be called the *enthalpy–pressure number* and links the enthalpy and pressure scales. It appears in front of the terms describing the effects of enthalpy additions and changes of composition in the volume and "converts" these effects into pressure changes.

$\Pi_1$ can be divided by $\Pi_{hp}$ to yield a form of the familiar enthalpy or phase–change number (Yadigaroglu and Bergles, 1972; Saha et al. 1976)

$$\Pi_{pch} = \frac{\Pi_1}{\Pi_{hp}} \equiv \frac{\dot{Q}^o}{\dot{j}^o \rho^o \Delta h^o} = \frac{\dot{Q}^o}{\dot{M}^o \Delta h^o} \tag{2.16}$$

where $\dot{M}^o \equiv \dot{j}^o \rho^o$ is a reference mass flow rate. We will see later that the latent heat is a natural scale for the reference enthalpy difference. Thus, the phase change number "converts" the heat additions into enthalpy differences.

The enthalpies $h_o$ appearing in the energy conservation equations (Equations 2.2 or 2.9) are *total* enthalpies (i.e., the sum of the intrinsic enthalpy of the fluid plus its kinetic and potential energies). Consequently, the exact scaling of these would have required separate consideration of enthalpy, velocity, and elevation scales. Since changes in kinetic and potential energy are very small or totally negligible, this complication can be avoided.

### 2.1.1 Case of a Perfect Gas

To gain some physical understanding regarding $f_2$, $e^*$ and $p^*$, consider the *case of a perfect gas*. The thermodynamic property $f_2$ scales the relative non–dimensional volume changes with pressure. For a perfect gas, $pv = RT$ and $R = c_p - c_v$, where $c_p$ and $c_v$ are the specific heats at constant pressure and volume, respectively, k their ratio, $c_p/c_v$, and R the perfect gas constant for the particular gas considered.

From the definition of $c_v$

$$e = c_v T = c_v \frac{RT}{R} = \frac{c_v}{c_p - c_v} pv = \frac{pv}{k - 1} \tag{2.17}$$

Thus

$$f_2 = \frac{\partial e/\partial p|_v}{v} = \frac{\frac{v}{k-1}}{v} = \frac{1}{k-1}$$

is in this case a constant. The property $e^*$ becomes

$$e^* \equiv e - v \left.\frac{\partial e}{\partial v}\right|_p = e - v \frac{p}{k-1} = e - e = 0$$

We realize that $e^*$ is not expected to attain very large values in real gas mixtures and $h_{o,i}$ should dominate the $(h_{o,i} - e^*)$ term. Thus, one sees that the first term on the right side of Equation 2.13 is not negligible and, to account properly for the effects of both enthalpy and heat additions to the system, *both* non–dimensional numbers appearing in Equation 2.13 (i.e., $\Pi_{hp}$ and $\Pi_1$) or, alternatively, the more familiar set $\Pi_{hp}$ and $\Pi_{pch}$, must be preserved.

Finally, for an ideal gas again, using Equation 2.17,

$$p^* = p + \left.\frac{\partial e}{\partial v}\right|_p = p + \frac{p}{k - 1} = \frac{k}{k - 1}\, p$$

From Equation 2.13 one notes that the effect of relative changes in volume is "amplified" by this ratio $p^*$ to produce relative changes in pressure.

### 2.1.2 Specific Frequencies of the Process

Another way of viewing the processes taking place is by considering their specific frequencies and time constants (Zuber, 1991).

Specific frequencies are given as ratios of a transfer intensity to capacity (amount) of the receiving volume (Zuber, 1991). In the particular case considered here, two specific frequencies involved are the ratio of heat addition $\dot{Q}^o$ and enthalpy addition $\dot{M}^o \Delta h^o$ to the heat capacity of the receiving volume $V^o \rho^o \Delta h^o$:

$$\omega_{\dot{Q}} \equiv \frac{\dot{Q}^o}{V^o \rho^o \Delta h^o}$$

and

$$\omega_{\Delta h} \equiv \frac{\dot{M}^o \Delta h^o}{V^o \rho^o \Delta h^o}$$

A *residence* or *fill* time $\tau^o$ has already been defined,

$$\tau^o \equiv \frac{V^o}{j^o} \tag{2.10}$$

The product of $\omega_{\dot{Q}}$ and $\tau^o$ results in the phase change number,

$$\omega_{\dot{Q}} \cdot \tau^o = \Pi_{pch}$$

as expected, while $\omega_{\Delta h} \cdot \tau^o = 1$; no new non-dimensional number is derived.

A third specific frequency is the ratio between the intensity of enthalpy addition $\dot{M}^o \Delta h^o$ and the "capacity of the volume to absorb work" $V^o \Delta p^o$:

$$\omega_{\Delta p} \equiv \frac{\dot{M}^o \Delta h^o}{V^o \Delta p^o}$$

The product of $\omega_{\Delta p}$ with $\tau^o$ produces, as expected, the enthalpy-pressure number

$\Pi_{hp}$,

$$\omega_{\Delta p} \cdot \tau^o = \Pi_{hp}$$

***In summary***, the preceding analysis revealed the presence of two non dimensional numbers, $\Pi_{hp}$ and $\Pi_{pch}$ and a time scale for the system, $\tau^o$ (Equations 2.15, 2.16, and 2.10, respectively). Identical values for the non–dimensional numbers will have to be maintained in the prototype and the model.

## 2.2 Phase Changes at Interfaces

The phase changes at interfaces involve the latent heat of vaporization and the interfacial mass flow rates and mass fluxes. These are considered in this section.

### 2.2.1 Latent Heat of Vaporization

An arbitrary enthalpy reference scale $\Delta h^o$ was used in the previous section. It is obvious that this arbitrary value could be chosen to coincide with the latent heat of the liquid used. Indeed, in systems with phase change, this seems to be the obvious choice. A simple confirmation of this fact will be provided here by considering mass continuity in a volume where phase change and mass transfers are taking place.

Figure 2–2 shows such a system consisting of the gas space with a mass M:

$$M = V \sum \rho_j$$

where $\rho_j$ are the partial densities of the constituents. For simplicity, consider a saturated mass of liquid vaporized by a heater providing power at the rate $\dot{Q}$. A mass flow rate $\dot{M}_{ex}$ leaves the vapor space of the system. Mass continuity for the vapor space results in:

$$\frac{dM}{dt} = \dot{M}_{LG} - \dot{M}_{ex} \tag{2.18}$$

where $\dot{M}_{LG}$ is the mass transfer rate by boiling given by

$$\dot{M}_{LG} = \frac{\dot{Q}}{h_{fg}} \tag{2.19}$$

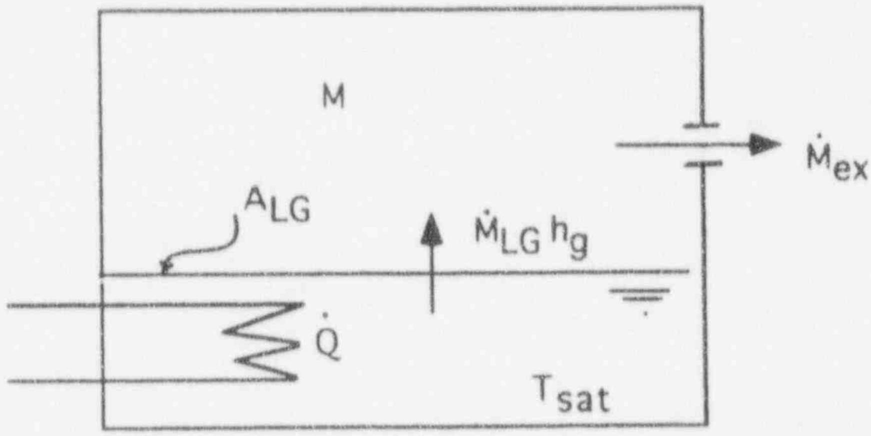



Figure 2-2 A volume containing a pool of boiling water.

where $h_{fg}$ is the latent heat of vaporization. Combining Equations 2.18 and 2.19 and non–dimensionalizing using $Q^o$, $V^o$, $j^o$, $\rho^o$, and $h^o_{fg}$ as scales, one obtains:

$$\frac{d}{dt^+}(V^+ \sum \rho_j^+) = \Pi'_{pch} \frac{\dot{Q}^+}{h^+_{fg}} - \dot{M}^+_{ex} \tag{2.20}$$

We note that an alternative phase change number

$$\Pi'_{pch} \equiv \frac{\dot{Q}^o}{j^o \rho^o h^o_{fg}} \tag{2.21}$$

has appeared naturally. Comparing the two–phase change numbers $\Pi_{pch}$ and $\Pi'_{pch}$ (Equations 2.16 and 2.21, respectively), and considering their ratio, it becomes evident that we should have identical ratios of $h^o_{fg}/\Delta h^o$ in the prototype and the model. The scale of enthalpies $\Delta h^o$ has not been specified so far; there is in principle no restriction for its choice. *When prototypical fluids under prototypical* *thermodynamic conditions are used in the model*, it becomes evident that a natural way of satisfying the identical $h^o_{fg}/\Delta h^o$ ratio requirement[a] is to take:

$$\Delta h^o = h^o_{fg} \tag{2.22}$$

In other words the latent heat of the fluid provides a natural scale of enthalpies. Thus we also define

$$\Pi'_{hp} \equiv \frac{h^o_{fg}}{\Delta p^o/\rho^o} \tag{2.23}$$

This form of the enthalpy–pressure number will be used instead of $\Pi_{hp}$ (Equation 2.15) in the following sections.

### 2.2.2 Rates of Phase Change

In the SBWR containment volumes, phase changes typically take place at the free surface of pools and on the walls. Condensation on structures and walls is limited by conduction within the structure and, therefore, depends strongly on the conduction characteristics of the walls. As already noted, conduction in the SBWR structures cannot easily be simulated by the experimental facilities (see Section 3.5 for details). It is left as an experimental parameter that must be addressed by measurement and detailed numerical calculations during data reduction.

In contrast, it is relatively straightforward to scale phase changes at the free pool surfaces. The flow rates due to phase change at the surface of a pool are given by the product of the pool surface area $A_{LG}$ times the mass flux due to phase change $\dot{m}_{LG}$. The latter, in general, may depend on the fluid conditions on both sides of the interface (p, T, partial densities of constituents $\rho_j$) and on hydrodynamic parameters controlling mass transfer (i.e., the Reynolds and Prandtl numbers of the fluids). The hydrodynamic dependence is considered in the bottom–up analysis of Section 3. Here we derive the scaling of the surface areas. Since the phase change effects were included in the convective enthalpy terms of Equation 2.7, (by now separating these and showing them explicitly), we get, instead of the first term on the right side of Equation 2.13, two terms (the second term could also be a *sum* of terms involving phase change at several surfaces):

[a] It is not necessary to deal here with the more complex cases involving use of a different fluid or of the same fluid but at a different pressure level for the model, since neither of these alternatives was retained for any of the SBWR related tests. The choice of an alternative modeling fluid or of a non–prototypical pressure level leads to much more complex and restrictive scaling laws than the ones derived here.

$$\frac{dp^+}{dt^+} = \frac{1}{f_2 V^+}\left[\ \Pi'_{hp} \sum\left[j_i^+ \rho_i^+ (h_{o,i}^+ - e^{*+})\right] + \frac{\tau^o}{\Delta p^o}\ \frac{A_{LG}\dot{m}_{LG}^o h_{fg}^o}{V^o}\ \dot{m}_{LG}^+ (h_g^+ - e^{*+})\ \right] + \ldots$$

We note a new non–dimensional number

$$\Pi_2 \equiv \frac{\tau^o}{\Delta p^o}\ \frac{A_{LG}\dot{m}_{LG}^o h_{fg}^o}{V^o}$$

Dividing $\Pi_2$ by $\Pi'_{hp}$ (Equation 2.23), we find a second, *interfacial* phase change number:

$$\Pi_{ipch} \equiv \frac{\Pi_2}{\Pi'_{hp}} = \frac{A_{LG}\dot{m}_{LG}^o}{j^o \rho^o} \tag{2.24}$$

This number is the ratio of evaporative to convective mass addition. It can be used to scale the pool interfacial areas.

### 2.2.3 Specific Frequency for Phase Change

Again, we can define a specific frequency for phase change as the intensity of phase change (rate of vaporization) divided by the amount (capacity) in the receiving gas volume:

$$\omega_{vap} \equiv \frac{A_{LG}\dot{m}_{LG}^o}{V^o \rho^o}$$

Multiplying by the *fill* time of the process

$$\tau^o \equiv \frac{V^o}{j^o} \tag{2.10}$$

we obtain the interfacial phase change number:

$$\omega_{vap}\ \tau^o = \Pi_{ipch}$$

***In summary***, the analysis of this section has motivated the choice of a reference latent heat $h_{fg}^o$ as *the* enthalpy scale to be used in the enthalpy–pressure and phase change numbers $\Pi'_{hp}$ and $\Pi'_{pch}$ (Equations 2.23 and 2.21) and yielded a new interfacial phase change number $\Pi_{ipch}$ (Equation 2.24).

## 2.3 Transfers of Mass Between Volumes Driven by Pressure Differences

Mass transfers between containment volumes are driven by pressure differences; these could be due to differential pressure buildup in two different volumes or may also have hydrostatic causes. In this section, we derive the similarity laws governing such pressure–difference–driven mass flow rates in channels (pipes, ducts, etc) connecting various containment volumes.

The isolation condenser tubes constitute part of such piping in the SBWR. In all the SBWR–related tests, the isolation condenser tubes have prototypical dimensions and operate under prototypical flow and pressure drop conditions. Since pressure differences are generally also preserved in all tests, there are no scaling considerations for the pressure drop in the condenser tubes. Consequently, only the case of adiabatic channels is considered here. This, together with the assumption of incompressible flow made below, allows integration of the momentum equation assuming constant density and thus simplifies the analysis. Any heat exchanges between the fluid in channels connecting two volumes and the fluid within the volumes traversed are clearly small and are considered on an ad–hoc basis in both system calculations and for the experimental data reduction.

The general case (Figure 2–3) of a pipe connecting two volumes at pressures $p_1$ and $p_2$ is considered. In the receiving volume, the pipe may be immersed in a pool of liquid; in this case, we call it a *vent*. We consider here the case of an "open" vent with flow discharging from the vent.

The case of *single–phase flow* between the two volumes is treated here since this is the case in most pipes connecting SBWR volumes. The flow is treated as incompressible since, at the flow rates and pressure differences considered here, the vapor can be treated as such. (This question is analyzed in more detail in Section 2.4.2.) The analysis leads to the definition of the characteristics of the piping in the model.

For pipes that may carry *two–phase flow*, the analysis would be similar but would involve, in addition, a two–phase frictional multiplier, which is a function essentially of the fluid properties and of flow quality, and, to a much lesser extent, of the pipe diameter, flow rate and other secondary variables. Since the tests will be conducted with prototypical fluid conditions, the condition of the fluid entering the pipes will also be prototypical. For the adiabatic (or nearly adiabatic, except for heat losses) cases considered here, the two–phase multiplier will thus depend only on identical or very similar inlet conditions. The effects of the other variables mentioned above are expected to be of second–order importance. Thus, the analysis presented below applies also to piping carrying a two–phase mixture.

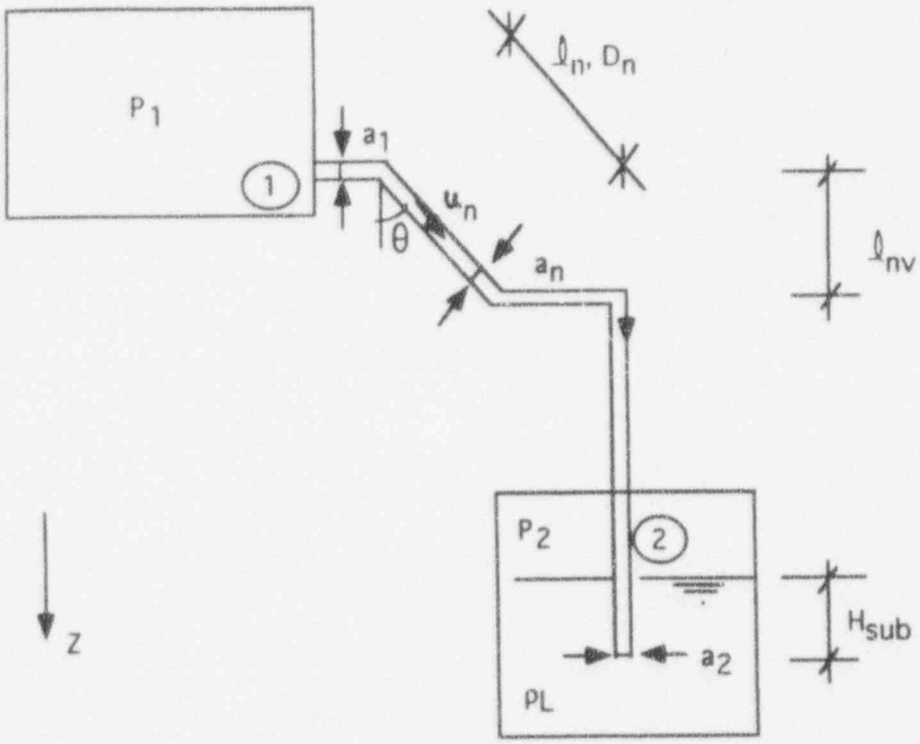



Figure 2–3 Pipe connecting two volumes and submerged in volume 2.

The one–dimensional momentum equation for time–dependent flow is:

$$\frac{\partial}{\partial t}(\rho u) + \frac{\partial}{\partial z}(\rho u^2) = -\frac{\partial p}{\partial z} + \rho g \cos\vartheta - \left[\frac{4f}{D} + k_n'\delta(z-z_n)\right]\frac{\rho u^2}{2}$$

The local losses at $z_n$ are considered by the terms $k_n'\delta(z - z_n)$, where $\delta$ is the Kronecker delta. This equation is integrated between points 1 and 2 for the piping system of Figure 2–3, consisting of a number of segments (identified by the subscript n) having flow areas $a_n$ and lengths $l_n$. The density $\rho$ is taken as constant, as discussed above. The integration results in

$$(p_2 - p_1) = -\rho\left(\sum \frac{a_r}{a_n} l_n\right)\frac{du_r}{dt} - \rho\left(\frac{a_r^2}{a_2^2} - \frac{a_r^2}{a_1^2}\right)u_r^2$$

$$+ \sum \rho g l_{vn} - \frac{\rho u_r^2}{2} \sum F_n \left(\frac{a_r}{a_n}\right)^2 - \rho_L g H_{sub} \tag{2.25}$$

where

$$F_n \equiv \frac{4 f_n l_n}{D_n} + k_n \tag{2.26}$$

The $k_n$ denote the local loss coefficients; the projections of the lengths of the various segments (having an angle of inclination $\vartheta$) on the vertical axis are denoted by $l_{vn}$; $H_{sub}$ is the submergence depth of the vent, and $a_r$ is a reference cross section used to eliminate the $u_n$ in favor of the reference velocity $u_r$. Indeed (for the case of constant–density flow considered here), from continuity:

$$a_n u_n = a_r u_r$$

The reference cross section $a_r$ is related to the reference volumetric flow rate by

$$j^o = u_r^o a_r$$

The second term of Equation 2.25 has the same form as the $F_n$ term and can be combined with it:

$$F \equiv \sum F_n \left(\frac{a_r^2}{a_n^2}\right) + 2\left(\frac{a_r^2}{a_2^2} - \frac{a_r^2}{a_1^2}\right) \tag{2.27}$$

The following parameters are also defined to simplify the notation:

The sum of the vertical projections of the various segments, $L_g$,

$$L_g \equiv \sum l_{vn} \tag{2.28}$$

and the equivalent *inertia* length of the piping, $L_I$,

$$L_I \equiv \sum \frac{a_r}{a_n} l_n \tag{2.29}$$

With these notations, Equation 2.25 takes the form

$$p_2 - p_1 = -\rho L_I \frac{du_r}{dt} + \rho g L_g - F \frac{\rho u_r^2}{2} - \rho_L g H_{sub} \tag{2.30}$$

Equation 2.30 is non-dimensionalized using as reference variables $\tau^o_{tp}$, $L_g$, $u^o_r$, $\rho^o$, $\Delta p^o$, and $H^o_{sub}$, for times, velocities, densities, pressure differences, and submergence, respectively. Its non-dimensional form is:

$$(p_2^+ - p_1^+) = -\left(\frac{\rho^o L_I u_r^o}{\Delta p^o \ \tau^o_{tp}}\right) \rho^+ \frac{du_r^+}{dt^+} + \left(\frac{\rho^o g L_g}{\Delta p^o}\right) \rho^+ - \left(\frac{F \rho^o u_r^{o2}}{\Delta p^o}\right) \frac{\rho^+ u_r^{+2}}{2} - \left(\frac{\rho_L g H^o_{sub}}{\Delta p^o}\right) H^+_{sub} \tag{2.31}$$

Four non-dimensional numbers have appeared; they are all cast as ratios of the various pressures to the reference pressure drop. These are

The *inertial pressure drop number*

$$\Pi_{in} \equiv \frac{\rho^o L_I u_r^o / \tau^o_{tp}}{\Delta p^o} \tag{2.32}$$

a *hydrostatic pressure number*,

$$\Pi_{hyd} \equiv \frac{\rho^o g L_g}{\Delta p^o} \tag{2.33}$$

a *pressure loss* number,

$$\Pi_{loss} \equiv \frac{F \rho^o u_r^{o2}}{\Delta p^o} \tag{2.34}$$

and the *submergence pressure drop number*,

$$\Pi_{sub} \equiv \frac{\rho_L g H^o_{sub}}{\Delta p^o} \tag{2.35}$$

The reference pipe-transit time scale $\tau^o_{tp}$ can be chosen to be the *pipe inertial characteristic time*

$$\tau^o_{in} \equiv \frac{L_I}{u^o_r} \tag{2.36}$$

Then $\Pi_{in}$ becomes

$$\Pi_{in} = \frac{\rho^o u_r^{o2}}{\Delta p^o} \tag{2.37}$$

Dividing $\Pi_{hyd}$ by $\Pi_{sub}$, we get

$$\frac{\Pi_{hyd}}{\Pi_{sub}} = \frac{\rho^o}{\rho_L}\frac{L_g}{H^o_{sub}} \tag{2.38}$$

Since for prototypical fluids the density ratio $\rho^o/\rho_L$ appearing in the equation above is preserved, we conclude that, instead of considering conservation of $\Pi_{hyd}$, it is sufficient to preserve the length ratio

$$\frac{L_g}{H^o_{sub}}$$

Finally, since

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

instead of $\Pi_{loss}$, it is sufficient to preserve $\Pi_{in}$ *and* F or their product.

### 2.3.1 Transit Times in the Piping

We consider now the specific frequency of the transfers of mass in the piping. Again, considering the ratio of an intensity of transfer (the volumetric flow rate) to the (volumetric) capacity of the piping, we obtain

$$\omega_{tr} \equiv \frac{j^o}{\sum l_n a_n} = \frac{j^o}{L_V a_r} \tag{2.39}$$

where the equivalent volume–length of the piping $L_V$ is

$$L_V \equiv \sum \frac{a_n}{a_r} l_n \tag{2.40}$$

The pipe transit time is the inverse of $\omega_{tr}$:

$$\tau^o_{tr} \equiv \frac{L_V}{u^o_r} \tag{2.41}$$

The product of $\omega_{pt}$ times the inertial characteristic time $\tau^o_{in}$ (Equation 2.36) produces a non-dimensional number relating inertial and transit times:

$$\omega_{tr}\tau^o_{in} = \frac{\tau^o_{in}}{\tau_{tr}} = \frac{j^o}{L_V a_r}\frac{L_I}{u^o_r}$$

Since

$$j^o = a_r u^o_r$$

we find that

$$\omega_{tr}\tau^o_{in} = \frac{\tau^o_{in}}{\tau_{tr}} = \frac{L_I}{L_V}$$

Note that this ratio is equal to

$$\frac{L_I}{L_V} = \frac{\sum \frac{a_r}{a_n} l_n}{\sum \frac{a_n}{a_r} l_n} \tag{2.42}$$

This produces another geometrical ratio that should be preserved.

***In summary***, the analysis of this section shows that we are left with $L_g/H^o_{sub}$, F (Equation 2.27) or the product $\Pi_{in}F$), the ratio $L_I/L_V$ (Equation 2.42), and only $\Pi_{in}$ and $\Pi_{sub}$ (Equations 2.37 and 2.35, respectively), as the non-dimensional quantities to match between prototype and experiment. Two additional time scales $\tau^o_{in}$ and $\tau^o_{tr}$, Equations 2.36 and 2.41, were also identified.

## 2.4 General Scaling Criteria

The criteria derived in Sections 2.1, 2.2, and 2.3 will be combined now to arrive at general scaling laws for the models of the SBWR.

Recall that the test facilities are designed to operate with prototypical fluids under prototypical thermodynamic conditions. In addition, the following **non-dimensional numbers** must be matched:

*Enthalpy-Pressure Number*

$$\Pi'_{hp} \equiv \frac{h^o_{fg}}{\Delta p^o/\rho^o} \tag{2.23}$$

*Phase–Change Number*

$$\Pi'_{pch} \equiv \frac{\dot{Q}^o}{j^o\rho^o h^o_{fg}} = \frac{\dot{Q}^o}{\dot{M}^o h^o_{fg}} \tag{2.21}$$

*Interfacial Phase Change Number*

$$\Pi_{ipch} \equiv \frac{A_{LG}\dot{m}^o_{LG}}{j^o\rho^o} = \Pi'_{pch}\,\frac{A_{LG}\dot{m}^o_{LG}h^o_{fg}}{\dot{Q}^o} \tag{2.24}$$

*Inertial Pressure Drop Number*

$$\Pi_{in} \equiv \frac{\rho^o u_r^{o2}}{\Delta p^o} \tag{2.37}$$

*Submergence Number*

$$\Pi_{sub} \equiv \frac{\rho_L g H^o_{sub}}{\Delta p^o} \tag{2.35}$$

In addition, we have defined three **time scales** that must be matched, namely $\tau^o$, $\tau^o_{in}$, and $\tau^o_{tr}$:

$$\tau^o = \frac{V^o}{j^o} \tag{2.10}$$

$$\tau^o_{in} \equiv \frac{L_I}{u^o_r} \tag{2.36}$$

$$\tau^o_{tr} \equiv \frac{L_V}{u^o_r} \tag{2.41}$$

and three **geometric parameters**,

$$\frac{L_g}{H^o_{sub}}$$

with $L_g$, the sum of the vertical projections of the piping segments defined by Equation 2.28; the ratio of the equivalent inertia and volume lenghts of the piping,

$$\frac{L_I}{L_V} = \frac{\sum \frac{a_r}{a_n} l_n}{\sum \frac{a_n}{a_r} l_n} \tag{2.42}$$

and the total flow resistance of the piping,

$$F \equiv \sum F_n \left(\frac{a_r^2}{a_n^2}\right) + 2\left(\frac{a_r^2}{a_2^2} - \frac{a_r^2}{a_1^2}\right) \tag{2.27}$$

where

$$F_n \equiv \frac{4 f_n l_n}{D_n} + k_n \tag{2.26}$$

As noted in Section 2.3, instead of preserving both F *and* $\Pi_{in}$, it is sufficient to preserve their product, $\Pi_{loss}$ (Equation 2.34).

### 2.4.1 Comparison of the Time Scales

The three time scales produced by the analysis of the previous sections ($\tau^o$, $\tau^o_{in}$, and $\tau^o_{tr}$) scale the rates of volume fill, of inertial effects, and of pipe transfers, respectively. Clearly, the systems considered here are made of large volumes connected by piping of much lesser volumetric capacity. The pressure drops between these volumes are not expected to be dominated by inertial effects. Thus the inertia and transit times, which are of the same order of magnitude, are much smaller than the volume fill times:

$$\tau^o \gg \tau^o_{in} \approx \tau^o_{tr}$$

We conclude that the time scale that "must" be preserved is $\tau^o$ and we write

$$\tau_R = 1 \tag{2.43}$$

where the subscript R denotes the ratio between the corresponding scales of prototype and model for the variable z:

$$z_R \equiv \frac{z_{prot}}{z_{mod}}$$

The other two time scales (controlled by the geometric characteristics $L_I$ and $L_V$ of the piping) are clearly of lesser importance.

In other words, we expect the time behavior of the system to be controlled by the volume fill rates (or $\tau^o$). The pipe transit times and the inertial time scale of the piping ($\tau^o_{tr}$ and $\tau^o_{in}$) are much shorter; the overall dynamics of the system are not controlled by such effects. Thus, in relation to the time constants of the system, *the lengths of piping connecting containment volumes and the velocities in these pipes do not have to be scaled exactly.*

### 2.4.2 Compressibility of the Gas Flowing in Pipes

The gases flowing in pipes connecting containment volumes were treated as incompressible; this assumption is justified in this section.

We start from the continuity equation, written for the pipe segment of Figure 2-4,

$$\frac{dM}{dt} = \dot{M}_1 - \dot{M}_2 \tag{2.44}$$

where $\dot{M}_1$ and $\dot{M}_2$ are the mass flow rates at Sections 1 and 2, respectively; in general

$$\dot{M} = A_p \rho u$$

M is the mass contained in the pipe of volume $V_p = A_p L_p$ and average density $\overline{\rho}$. We non-dimensionalize Equation 2.44 by defining

$$\dot{M}^+ \equiv \frac{\dot{M}}{\dot{M}^o_1}$$

$$\rho^+ \equiv \frac{\rho}{\rho^o_1}$$

and a pipe transit time

$$\tau^o_{tr} \equiv \frac{L_p}{u^o_1}$$

Equation 2.44 takes the non-dimensional form

$$\tau_{tr}^{o} \frac{d}{dt}\overline{\rho^{+}} = \dot{M}_{1}^{+} - \dot{M}_{2}^{+} \tag{2.45}$$

It is evident that if $\tau_{tr}^{o}$ and the rate of change of the average density are both small, the mass flow rates at the inlet and the exit of the pipe will be approximately equal, $\dot{M}_1^+ \approx \dot{M}_2^+$, or $\dot{M}_1 \approx \dot{M}_2$. Clearly, the pipe transit time $\tau_{tr}^{o}$ must be compared to the other time constants of the system; namely, the ones determining the variation of the conditions in the containment volumes (i.e., $\tau^{o}$). The same volume fill constant $\tau^{o}$ determines the rate of variation of the inlet density $\rho_1$ and, consequently, of the average density $\overline{\rho}$ in the pipe.

### 2.4.3 Length Scales of the System

The three geometric parameters that have appeared (i.e., $L_g/H_{sub}^{o}$, $L_I/L_V$ and F) are considered now.

In facilities preserving vertical heights, the ratio $L_g/H_{sub}^{o}$ is clearly conserved. The ratio $L_I/L_V$ scaling inertial to transit times in the piping should not be very important, according to the discussion of Section 2.4.1. It can, however, approximately at least, be conserved. Thus, we require:

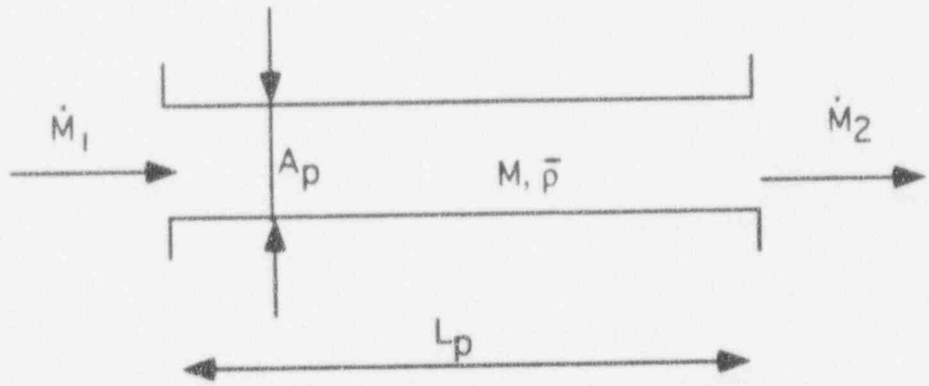



Figure 2-4 A pipe segment connecting two volumes.

$$\left(\frac{L_g}{H_{sub}}\right)_R = \left(\frac{L_I}{L_V}\right)_R = 1 \tag{2.46}$$

The factor F, (Equation 2.27) determining the total flow resistance of the piping is important. This factor does not have to be scaled alone, however, but rather as the product $\Pi_{in}F$, as shown in Section 2.3. The scaling of the pressure losses in the piping is considered in Section 2.4.5 below. Since the F values of the models can in general be larger than those of the prototypes (because of the smaller diameters in the ratio $fl/D$), the velocities in the model may end up being smaller than those of the prototype. This is not important as long as the transit times between volumes are small compared to the volume fill times $\tau^o$, as discussed in Section 2.4.1.

### 2.4.4 Other Reference Scales in the System

We have already dealt with the time and geometric scales in the preceding two paragraphs. We can consider now the remaining five non-dimensional numbers listed at the beginning of Section 2.4 and repeated here for convenience:

*Enthalpy-Pressure Number*
$$\Pi'_{hp} \equiv \frac{h^o_{fg}}{\Delta p^o/\rho^o} \tag{2.23}$$

*Phase-Change Number*
$$\Pi'_{pch} \equiv \frac{\dot{Q}^o}{j^o \rho^o h^o_{fg}} \tag{2.21}$$

*Interfacial Phase Change Number*
$$\Pi_{ipch} \equiv \frac{A_{LG}\dot{m}^o_{LG}}{j^o \rho^o} \tag{2.24}$$

*Inertial Pressure Drop Number*
$$\Pi_{in} \equiv \frac{\rho^o u_r^{o2}}{\Delta p^o} \tag{2.37}$$

*Submergence Number*
$$\Pi_{sub} \equiv \frac{\rho_L g H^o_{sub}}{\Delta p^o} \tag{2.35}$$

and the time scale $\tau^o$,

$$\tau^o = \frac{V^o}{j^o} \tag{2.10}$$

To achieve proper scaling, the five non-dimensional numbers listed above must have the *same values for each pair of system and model components* (the values that a non–dimensional number takes will of course vary between pairs of components). This leads to the definition of the proper scaling ratios between prototype and model. For example, conservation of the enthalpy–pressure number everywhere in the system leads to the scaling requirement:

$$\left(\frac{h_{fg}^{o}}{\Delta p^{o}/\rho^{o}}\right)_R = 1$$

for all system components[b].

The minimum set of independent reference scales appearing in the non–dimensional numbers listed above is:

$$h_{fg}^{o},\ \Delta p^{o},\ \rho^{o},\ \dot{Q}^{o},\ j^{o},\ A_{LG},\ \dot{m}_{LG}^{o},\ u_r^{o},\ \rho_L,\ H_{sub}^{o},\ \text{and}\ V^{o}$$

The reference scales can be chosen uniquely for the entire system or, alternatively, for each system component. This does not make any difference, since it is only ratios of scales that must be compared between prototype and model; these ratios are the ones denoted by the subscript R, as shown below.

When prototypical fluids are used, $h_{fg}^{o}$, $\rho^{o}$, $\rho_L$, and $\dot{m}_{LG}^{o}$ need not to be considered. (Some reservations were already made regarding the conservation of the phase change flux $\dot{m}_{LG}^{o}$ in Section 2.2.2; these are discussed in Section 3.) Thus, we are left with only

$$\Delta p^{o},\ \dot{Q}^{o},\ j^{o},\ A_{LG},\ u_r^{o},\ H_{sub}^{o},\ \text{and}\ V^{o}$$

We are left with seven independent scales and only five non–dimensional numbers, plus an arbitrary "system scale" to be determined. However, no choice for a $\Delta p^{o}$ scale has been made up to this point. The submergence hydrostatic head $\rho_L g H_{sub}^{o}$ is an important parameter, since it largely controls the flows of mass and energy between the SBWR containment volumes. Thus it appears to be the natural choice for the (so far, arbitrary) value of $\Delta p^{o}$. By dictating the use of the submergence hydrostatic head as *the* reference pressure drop, the submergence number $\Pi_{sub}$ (Equation 2.35) takes the value:

$$\Pi_{sub} = 1$$

[b] The subscript R, already defined in Section 2.4.1, denotes the ratio between corresponding quantities in the prototype and the model.

in both prototype and model, and

$$\Delta p^o = \rho_L g H^o_{sub} \tag{2.47}$$

Thus, all pressure drops will scale with $\rho_L g H^o_{sub}$, and the submergence number $\Pi_{sub}$ needs no longer to be considered. Insertion of Equation 2.47 in the enthalpy–pressure number, Equation 2.23, yields:

$$\Pi'_{hp} = \frac{\rho^o}{\rho_L} \frac{h^o_{fg}}{gH^o_{sub}} \tag{2.48}$$

Since the fluid is prototypical, $(\rho^o h^o_{fg}/\rho_L)_R = 1$, and Equation 2.48 shows that the *submergence depths must be conserved* at a scale of 1:1,

$$(H_{sub})_R = 1 \tag{2.49}$$

Otherwise, the rates of pressure change due to thermodynamic evolutions (considered in Section 2.1) will not match the pressure differences driving the mass and energy transfers between volumes (considered in Section 2.3).

We are left now with five scales:

$$\dot{Q}^o,\ \dot{J}^o,\ A_{LG},\ u^o_r, \text{ and } V^o$$

and three non–dimensional numbers:

$$\Pi'_{pch},\ \Pi_{ipch}, \text{ and } \Pi_{in}$$

Conservation of the phase change number $\Pi'_{pch}$ dictates the need to preserve the ratio $\dot{Q}^o/\dot{J}^o$. Similarly, conservation of the interfacial phase change number, $\Pi_{ipch}$, requires preservation of the ratio $A_{LG}/\dot{J}^o$. Clearly, this can only be achieved by

$$\dot{Q}_R = \dot{J}_R\ (= \dot{M}_R) = (A_{LG})_R = R \tag{2.50}$$

where R is the "system scale". Thus, heat addition, flow rates, and horizontal areas must scale with the system scale R. Since the volume scale $V^o$ appears only in the time constant $\tau^o$, one could in principle conduct tests at a different time scale (not 1:1) by modifying the volume scale $V_R$. This is possible as long as $\tau^o$ is *the* controlling time scale, as already discussed in Section 2.4.1. Accelerated tests can, for instance, be conducted by decreasing $V_R$ or increasing equally the other scales, $\dot{Q}_R = \dot{M}_R = \dot{J}_R = (A_{LG})_R$. Conservation of the time scale $\tau^o$ also implies (in any case) preservation of the ratio $V^o/\dot{j}^o$.

### 2.4.5 Scaling of the Piping

Scaling of the pressure drops between compartments requires consideration of the product $F\Pi_{in}$, as shown in Section 2.3. Since $\Delta p_R = (\rho_L g H_{sub})_R = 1$, this amounts to

$$\left(u_r^2 F\right)_R = 1$$

Since

$$u_r^o = \frac{\dot{j}^o}{a_r}$$

we obtain

$$\left(a_r^2\right)_R = \left(F\dot{j}^2\right)_R$$

or

$$(a_r)_R = (F^{\frac{1}{2}}\ \dot{j})_R \tag{2.51}$$

This equation determines the reference flow area scale $(a_r)_R$. The factor F, Equations 2.26 and 2.27, depends on both the frictional losses in the pipes, i.e. on the groups $4f_n l_n/D_n$, and on the form losses $k_n$. The latter are generally insensitive to scale. Since the model diameters $D_n$ are smaller, however, the F factors of the models tend to be larger. Thus Equation 2.51 leads to an increase of the model pipe diameters; this reduces the flow velocities.

***In practice***, pipe scaling is performed according to the following procedure: the pipe cross–sectional areas in the scaled facilities are oversized for convenience; this leads to somewhat lower flow velocities in the pipes. Thus, considering only the *form losses* (for which the loss coefficients are only weakly dependent on flow velocity or Reynolds number), the total $\Delta p$'s in the models would be *lower* than prototypical. On the contrary, *wall friction* in the scaled facilities is *larger* (due to larger values of the $fl/D$ values produced by the smaller pipe diameters), as it cannot be compensated in general by the decrease in velocity. Usually (and fortunately), the total pressure drops in the piping are dominated by form losses, so that the total $\Delta p$'s in the scaled facilities end up being somewhat smaller. They can therefore be matched by introducing additional form losses by local orificing.

The pipe flow areas determined in this fashion result in velocities that do not lead to matching **pipe transit times**, as expected. Indeed, using Equation 2.51, the velocity scaling ratios are given as:

$$(u_r)_R = \left(\frac{\dot{J}}{a_r}\right)_R = (F^{-\frac{1}{2}})_R \tag{2.52}$$

As already noted, since the F values of the models are in general larger than those of the prototypes, the velocities in the model are generally smaller than in the prototype. This is not important as long as the transit times between volumes, $\tau^o_{tr}$ (Equation 2.41) are small compared to the volume fill times $\tau^o$, as discussed in Section 2.4.1, and as long as extremely low velocities do not introduce new phenomena in the models.

The matching of the pressure drops in the various facilities is again discussed in Section 4: in summary, *matching of the total pressure drops is accomplished by using orifices in conjunction with convenient choices for pipe diameters.*

## 2.5 Summary

The analysis presented in this section has shown that, when prototypical fluids under prototypical thermodynamic conditions are used:

- The elevations in the prototype and in the model must be identical, especially the submergence depths of the vents.
- The volumetric flow rates, heat inputs and horizontal pool areas must be scaled according to the system scale R,

$$\dot{Q}_R = \dot{J}_R \ (= \dot{M}_R) = (A_{LG})_R = R \tag{2.50}$$

- If, in addition, the volumes are also scaled with R,

$$V_R = R$$

  the time scale between model and prototype is 1:1. In this case, we can speak of a vertical *slice* or vertical *section* model of the prototype.

- The pipe flow areas must be geometrically similar and the reference cross–sectional areas $a_r$ must scale like

$$(a_r)_R = (F^{\frac{1}{2}} \dot{J})_R \tag{2.51}$$

where the factor F, determining the total pipe losses, can be adjusted by introducing local losses in the model to match the pressure drops, if necessary.

– The pipe flow areas determined in this fashion result in velocities that do not match the pipe transit times; usually, the velocities in the model may be smaller than those of the prototype. This is, however, not important as long as the transit times between volumes are small compared to the volume fill times $\tau^o$.

# 3. Scaling of Specific Phenomena – Bottom–Up Approach

The scaling of particular SBWR system components in relation to specific phenomena and processes considered important is conducted in a bottom–up fashion in this section. The discussion is limited to the spatial–scale–dependent phenomena ranked as important in the SBWR PIRT and not considered generically in Section 2.

## 3.1 Important Phenomena

The SBWR PIRT was used to identify the phenomena of safety importance for post–LOCA behavior of the SBWR. The important phenomena for each subsystem or component and for each phase of the class of accidents considered were identified. Table 3–1 lists all phenomena that received importance grades of 7, 8, or 9 on a scale of 0 to 9. The phenomena (in each of the three main phases of the class of accidents considered) are listed, together with the subsystems where they are expected to be of importance.

The last column of Table 3–1 shows how the scaling issue for each phenomenon was addressed. In several cases (e.g., "friction"), the scaling concern was addressed generically in the top–down scaling analysis of Section 2. Such phenomena are marked "top–down." In the case of condensation phenomena within the condenser tubes, scaling is addressed by the use of full–size tubes in system test facilities, supported by the single–tube tests at MIT and UCB. The University tests are summarized in Section 3.6.2. Detailed "bottom–up" scaling is provided in the following sections for the remaining phenomena identified in Table 3–1. In these cases the number of the particular section where the scaling is addressed is shown in the last column of the table.

## 3.2 Thermal Plumes, Mixing, and Stratification

Thermal plumes, mixing and thermal stratification phenomena can be encountered:

- In the DW and in the gas space of the SC, for steam and noncondensable gases (nitrogen or hydrogen).
- In the suppression pool.

Combinations of single–phase/two–phase, axisymmetric/plane, and free/wall plumes

**Table 3–1**
**Important Phenomena Identified in PIRT**

| PHENOMENA/PROCESSES | LOCATION IN SYSTEM | SCALING |
|---|---|---|
| **Late Blowdown Phase (of importance for GIST tests)** | | |
| Critical flow | break, main vents, SRV quenchers, DPV | top–down[a] |
| Friction | break, main vents, SRV quenchers, DPV, IC/PCC lines | top–down |
| Void fraction/interfacial shear | main vents, SRV quenchers, DPV | top–down |
| Phase separation/interfacial shear | DW, SC(SP) | top–down/Sec. 3.2 |
| Liquid entrainment | break, main vents, SRV quenchers, DPV | top–down |
| Entrainment in jets | SC (SP: gas/liquid) | Sec. 3.5 |
| Component separation/mixing | DW (gases) | Sec. 3.2.3 |
| Flashing/evaporation | DW | top–down |
| Interfacial heat transfer/condensation | DW, SC(main + PCC vents, SRV quencher) | top–down/Sec. 3.5.2 |
| Degradation of condensation | SC(gas space), SC(main + PCC vents) | top–down/Sec. 3.5.2 |
| Condensation in tubes | IC, PCC | Univ. tests/Sec. 3.6.1 |
| Degradation of condensation in tubes | PCC | Univ. tests/Sec. 3.6.1 |
| Shear–enhanced condensation | IC tubes, PCC tubes | Univ. tests/Sec. 3.6.1 |
| **GDCS Phase (of importance for GIST, GIRAFFE, and PANDA tests)** | | |
| Friction | GDCS injection line | top–down |
| Condensation in tubes | IC, PCC | Univ. tests/Sec. 3.6.1 |
| Degradation of condensation in tubes | PCC | Univ. tests/Sec. 3.6.1 |
| Shear–enhanced condensation | IC tubes, PCC tubes | Univ. tests/Sec. 3.6.1 |
| **Long–Term Cooling Phase (of importance for GIRAFFE and PANDA tests)** | | |
| friction | PCC lines | top–down |
| Phase separation/interfacial shear | SC(gas space) | top–down, Section 3.2.2 |
| Component separation/mixing | DW, SC(gas space) | Sec. 3.2.3 |
| Mixing/entrainment into jets | DW (gases), SC(SP: gas/liquid) | Sec. 3.2, 3.5 |
| Buoyancy/natural circulation | DW, SC(SP), IC pools | Sec. 3.2, 3.6.2 |
| Forced flow | PCC fan | top–down |
| Interfacial heat transfer/condensation | SC, PCC vents, pool surfaces, containment spray | Sec. 3.3, 3.5.2 |
| Degradation of condensation (n/c's) | SC, PCC vents, pool surfaces, containment spray | Sec. 3.3, 3.5.2 |
| Condensation in tubes | PCC | Univ.tests/Sec. 3.6.1 |
| Degradation condensation in tubes | PCC | Univ.tests/Sec. 3.6.1 |
| Shear enhanced condensation | PCC tubes | Univ.tests/Sec. 3.6.1 |
| Lateral entrainment in 2–phase flow | condensers in IC pool | Sec. 3.6.2 |
| Component separation | PCC tubes | Univ. tests |
| Conduction in walls/int'nal structures | DW, SC | Not scaled/Sec. 3.4 |
| Steam bypass/leakage | DW–SC | Top–down |

[a] reduced to correct definition of the choked area

for fluids emerging from vents or originating on hot or cold wall surfaces can be encountered. The various stratification, plume, and jet situations are sketched in Figure 3–1.

The situations involving mixing induced by plumes are discussed in this section, while the condensation phenomena from either jets or two–phase plumes are considered in Section 3.5.2.

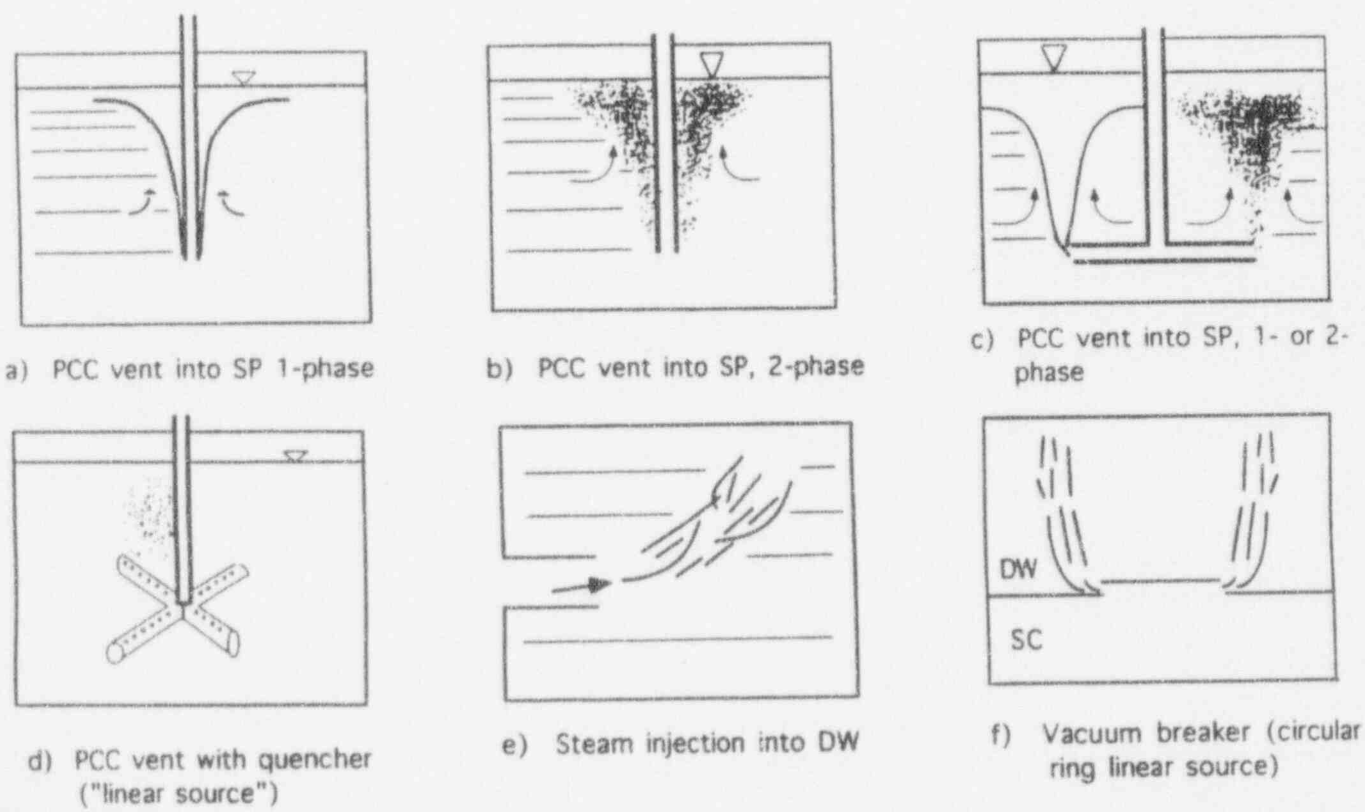



Figure 3–1 Thermal plumes and jets, and associated mixing and stratification phenomena in the SBWR.

### 3.2.1 Stratification and Mixing of the Suppression Pool

Possible stratification of the suppression pool is an important issue, since the temperature of the top layer of pool water determines the saturation pressure of the vapor in the gas space of the SC which, together with the partial pressure of the noncondensable gas, determines the containment pressure level.

The PCC vents inject noncondensable gases and steam into the SP at temperatures somewhat in excess of the SP water. Ideally, the steam condenses near the injection point. (Other possible situations, such as partial channeling of steam to the SP gas volume, are discussed in Section 3.5.2.) The SP may become stratified during the long term containment cooling phase, since the hot gases and the hot condensate will create plumes that will rise to the surface of the pool and spread horizontally.

#### 3.2.1.1 Horizontal Spreading

The horizontal spreading of the hot plume on the pool surface takes place at a velocity having as order of magnitude the velocity of a gravity wave (e.g., Moody, 1990) given by

$$\sqrt{gH}$$

where H is the height of the "film" of hot water spreading on the colder pool surface. For a hot water layer of *only* 1 mm thickness, we obtain a horizontal spreading velocity of the order of 0.1 m/s. Thus, within a few tens of seconds at most, the hot water spreads on top of the pool up to the walls. This time is short compared to the time scale of containment response and *the horizontal spreading of the plume can be considered as being instantaneous*[b].

Consequently, at any instant, *the surface of the pool will have a temperature equal to the average plume temperature reaching the surface of the pool.* That temperature will depend on the dilution of the initial mass injected by the rate of entrainment of liquid into the plume from the colder pool (i.e., on plume behavior only).

#### 3.2.1.2 Vertical Stratification

The layers of hot water spread on the surface of the pool will be displaced downwards by subsequent, hotter layers spreading on the surface (Smith et al.,

[b] This statement can be verified in the PANDA test facility where the hot plume rising from a vent in one of the SC vessels can spread towards the nearest vessel wall and also cross the large pipe connecting the two SC vessels and propagate in the second vessel, traversing a much larger distance, comparable to the circumferential distance between vents in the SBWR SC.

1992). This process will produce a degree of pool stratification, dependent, of course, on the amount of entrainment into the plume from the surrounding pool. With sufficiently large entrainment, the liquid reaching the surface of the pool will be only slightly above the pool average temperature, and the pool will be well mixed above the injection point.

If the entrainment into the plume is scaled properly (i.e., if the plume reaching the surface of the pool has the correct temperature history), it is evident that the correct scaling of the stratification (i.e., identity of the temperature gradients in a vertically 1:1 scaled facility or, alternatively, identical downward displacement velocity of the stratified fluid front) requires pool surface area to volumetric flow rate scaling:

$$(A_{LG})_R = \dot{J}_R = \dot{Q}_R = R$$

This condition already resulted from the top–down scaling considerations of Section 2. Thus, we will concentrate now on scaling of plume behavior.

### 3.2.2 Scaling of Plumes in Suppression Pool

Free plumes can be classified as laminar or turbulent. Geometrically, one distinguishes between axisymmetric round plumes and plane plumes. Thus, four different combinations exist (Gebhart et al., 1988). The scaling of plumes was recently discussed in relation to the SBWR by Peterson et al. (1993). Wall plumes (i.e., plumes rising around pipes or other vertical walls) provide lesser entrainment than free plumes (ibid) and should be avoided, if good mixing is desired[c]. Simple vertical pipe vents located near the vessel wall were, however, used in the GIRAFFE facility. Such vents will create wall plumes; this question is further discussed in Section 4.2.

The scaling of fully–developed plumes (i.e., plumes having self–similar radial distributions at various elevations), is relatively straightforward. The discussion of this section applies to such plumes. It is also assumed that the plumes do not interact with vessel walls and with neighboring plumes. This will be the case in sufficiently large–scale experiments.

In the following discussion, we consider first a **free, round, turbulent, buoyancy driven plume.** According to List (1982), at a given height z, the plume volume flow rate $\dot{J}(z)$ is given in this case by

[c] The PCC vents will be terminated by either horizontal quenchers distributing the gases essentially as a plane plume, or by a number of nozzles providing multiple free round plumes.

$$\dot{j} = k_{\mu}\ B^{\frac{1}{3}}\ z^{\frac{5}{3}} \tag{3.1}$$

where B is the specific buoyancy flux

$$B \equiv g\ \frac{\rho_a - \rho_o}{\rho_o}\ \dot{j}_o \tag{3.2}$$

$k_{\mu}$ a coefficient, and $\dot{j}_o$ the volumetric injection rate of mass at a density $\rho_o$, different from the ambient pool density $\rho_a$. In the case of steam condensing at the exit of the vent, $\dot{j}_o$ would be the resulting volumetric flow rate of condensate at temperature $T_o$ injected into the pool at temperature $T_a$. The energy brought into the pool by the temperature increment $T_o - T_a$ will remain in the plume. The average temperature of the plume at an elevation z, T(z) can be obtained from an energy balance between the injection point and z:

$$\dot{j}_o(T_o - T_a) = \dot{j}(z)(T(z) - T_a) \tag{3.3}$$

Combining Equations 3.1 through 3.3, one obtains[d]

$$\frac{T(z) - T_a}{T_o - T_a} = \frac{\dot{j}_o}{\dot{j}(z)} = \frac{1}{k_{\mu}\left[g\dfrac{\rho_a - \rho_o}{\rho_o}\right]^{1/3}}\ \frac{\dot{j}_o^{2/3}}{z^{5/3}} \tag{3.4}$$

For prototypical fluids and identical temperatures between prototype and model, one concludes that

$$(T(z) - T_a)_R = \left(\frac{\dot{j}_o^{2/3}}{z^{5/3}}\right)_R \tag{3.5}$$

Thus, one finds out that when the vertical elevations are preserved,

$$(T(z) - T_a)_R = (\dot{j}_o)_R^{2/3} = R^{2/3} \tag{3.6}$$

and that, in general, the temperature of the plume reaching the surface of the pool will not scale properly, since $(T(z) - T_a)_R$ in the model will be smaller by a factor $R^{2/3}$ than in the prototype instead of being equal. Physically, this is mainly due to the fact that we are dealing with the behavior of a *point* source that is much weaker in the model. The buoyancy flux injected *at a point* fully determines plume

[d] The same dependence on $\dot{j}$ and z can be obtained following the scaling laws provided by Chen and Rodi (1980), Chapter 4.

behavior and cannot be scaled (at least when submergence is maintained): the lower buoyancy and entrainment provided by a weaker point source are only partially compensated by the corresponding reduction in the heat input.

However, it is most likely that the vent design of the prototype will spread the gases into the pool via a number of small vents or a quencher, producing essentially a **plane plume from a line source** by a linear arrangement of multiple nozzles. According to Equation 3.4, distribution of the gases in N nozzles, each having a volumetric flow rate $\dot{J}/N$, will reduce $T(z) - T_a$ by a factor $N^{2/3}$ and essentially promote mixing.

In this case, perfect scaling of stratification of the pool can be achieved by choosing

$$N_R = \dot{Q}_R = \dot{J}_R = R \qquad (3.7)$$

In essence, this produces a scaled number of identical plumes in the prototype and the model.

In the case of a **linear quencher**, perfect scaling can be achieved by including in the model a scaled fraction of the total length of the quencher in the prototype. This produces in the model an identical segment of the prototype plume. In this case, the length of the quencher in the model should be scaled down by the system scale R.

If a design having **a few large vents** only is retained for the SBWR, there will be some unavoidable scaling distortion regarding pool stratification, according to Equation 3.4, unless one attempts to modify the plume height in order to have

$$(T(z) - T_a)_R = \left(\frac{\dot{J}_o^{2/3}}{z^{5/3}}\right)_R = 1 \qquad (3.8)$$

or

$$(H_{sub})_R = (\dot{J})_R^{2/5} \qquad (3.9)$$

It is unlikely, however, that good simulation will result since, for short plumes, the similarity conditions upon which the theories used above rely break down. The behavior of relatively short plumes is strongly affected by their initial developing region in a way which is difficult to analyze and scale. We have already seen in Section 2 that modifying the submergence depths distorts the thermodynamic behavior of the system. If the unlikely case that a few large vents are retained for the SBWR, this scaling difficulty could be resolved by conducting two series of experiments covering submergence depths meeting *both* requirements. In one series of such experiments, the emphasis will be on pool stratification and in the second, on containment thermodynamics. Thus, a system code could be qualified regarding both aspects of the problem. This may not be required, however, if there is no significant stratification in the system due to efficient mixing by the vent plumes, as discussed below.

A variety of situations can be encountered, depending on the rate of injection of steam and noncondensables from the PCC vents. When the steam injection rates are high, one would expect stratification, in spite of better mixing of the pool by the higher entrainment rate, because of the larger source term, as shown by Equation 3.4. However, any amount of noncondensables present in the steam will strongly promote mixing by creating two-phase plumes. Two-phase bubbly plumes can be shown to be much more effective in mixing the pool than single-phase plumes (Coddington, 1993a). Information on entrainment in bubbly plumes is provided by several authors: Fannelop and Sjoen (1980) and Milgram (1983) also review previous work. Relevant information on the behavior of two-phase plumes related to aerosol pool scrubbing experiments was reported at a specialized meeting (Huebner, 1985). The directly related question of condensation of steam containing noncondensables injected from the PCCS vents is discussed in Section 3.5.2. In any case, the use of a scaled length of the prototype quencher in the test facility will lead to correct scaling of the mixing process.

### 3.2.3 Stratification and Mixing of Gases in the Drywell

A situation similar to the one presented in the previous section for plumes in the SP arises regarding hot or cold and/or steam or noncondensable-gas plumes in the DW. The geometry of the DW is relatively complex and the plumes can interact with structures, walls, etc. Releases from breaks in the primary system will create hot plumes or jets of steam; vacuum breaker openings will introduce plumes or jets of gases from the SC into the DW. Differences in the temperatures of vertical surfaces in the DW can produce rising hot and descending cold wall plumes; free plumes can be created by evaporation from the surface of pools of water. In relatively simple geometries, the correct scaling of such phenomena will be possible as long as the situation can be characterized by identical plumes or segments of linear plumes formed from nozzles or jets, both in the prototype and the models; this is the case of plumes created by a relatively large number of injection points or by linear sources.

For free round plumes, there is also the possibility of modifying the height of the fluid above the injection point and/or the volumetric flow rate per injection point to provide for correct scaling, according to Equation 3.5. The same dependence of plume behavior on height and volumetric flow rate was obtained by Peterson et al. (1993), who also consider both buoyant jets and plumes, as well as the transition points between laminar and turbulent situations; consideration of the latter is also required for correct simulation of the plumes.

The elevations of the injection points could not be modified in the case of plumes in the SP considered in the previous section, since submergence of the vents had to be preserved. The absence of submergence preservation constraints in the DW opens the interesting possibility of modifying the elevation of the injection points in the models to better meet the scaling criteria using scaling criteria similar to the ones presented by Peterson et al. (1993); such scaling must be performed on a case-by-case basis.

The primary system **steam injection** situations that can be encountered in the DW of the SBWR are necessarily very diverse. The scaled experiments can also provide information about limiting envelope situations that can be used for code asessment.

The **vacuum breakers** can be visualized as horizontal disks, having a diameter of the order of 0.5 m, lifting under the pressure difference between the SC and the DW. Plumes from the vacuum breakers will likely inject gases at an angle close to the horizontal from the circular rim of these disks. Thus, to a good approximation, the jets/plumes from the vacuum breakers can be considered as linear sources having as length the perimeter of the vacuum breaker (Figure 3-1f). Correct scaling can be achieved by having in the models the scaled fraction of this perimeter.

## 3.3 Heat and Mass Transfers at Liquid-Gas Interfaces

Heat and mass transfers at liquid-gas interfaces (such as the surface of pools and of liquid films draining along the walls) depend on the interfacial surface area and on the variables driving the exchanges; namely, the state of the fluids at the interface and the hydrodynamic condition (i.e., the fluid velocities) near the interface.

The scaling of the **horizontal interfacial surface areas** was considered in Sections 2.2.2 and 2.4.3. The horizontal interfacial areas (e.g., pool surfaces) can be made to correctly scale with the system scale: $(A_{LG})_R = R$. The fluids used in the experiments and their thermodynamic states are prototypical. Thus, regarding mass transfers at horizontal interfacial areas, the only remaining scaling issue is the effect of the hydrodynamic conditions, i.e. essentially of the fluid velocities near the interfaces. This question is examined in Section 3.3.1 below.

The situation is different for **vertical interfacial areas** such as liquid films on the walls. Phenomena taking place at vertical surface areas cannot be scaled accurately, since these areas cannot be scaled down exactly. More important, the heat transfers into the walls, which are often driving the interfacial mass transfers (e.g., condensation on liquid films along the walls), cannot be scaled either, due to the widely differing conduction heat transfer characteristics of the structures. However, they can be accurately estimated (see Section 3.4). Thus, we should only make sure that the vertical–interface phenomena taking place in the SBWR and its models are of similar orders of magnitude in relation to the heat and mass transfer phenomena which dominate *overall* system behavior. The data obtained from the scaled experiments can then be used to qualify the system code.

### 3.3.1 Interfacial Transfers at Horizontal Surfaces

The state of the fluids in the models being essentially prototypical, the temperature and concentration differences driving the interfacial exchanges should be very similar in the prototype and the models. The remaining question raised here is the effect of the flow conditions in the proximity of the interface on the heat and mass exchange *coefficients*.

The rates of condensation or evaporation on horizontal pool surfaces will be limited by diffusion of noncondensables or sensible heat transfer on the gas side and by heat transfer on the liquid side. No rapid circulation of either the water or the gases near the pool surface is expected (except in the presence of strong plumes reaching the surface, as discussed below). In most situations we may encounter rather stagnant flow conditions. If there is flow (on either side) parallel to the interface, the heat and mass exchange coefficients will depend on some fractional power of the Reynolds number (i.e., of the velocity of the fluids). It is not possible to make *general* scaling statements regarding such flows and their influence on the interfacial exchanges. No strong effects should be expected, however, unless some particularity of the design creates locally high velocities. Such questions should be examined case–by–case for all the facilities involved.

Strong single–phase or two–phase plumes reaching the surface of pools will spread horizontally and produce significant movement of the surface water. In this case the exchanges between the gas and the liquid spaces will be enhanced. Such phenomena will be properly scaled if the plumes themselves are scaled adequately (see Sections 3.2 and 3.5).

## 3.4 Heat Capacity of Containment Structures and Heat Losses

The walls and structures of the SBWR containment are complex composite structures with very large **heat capacity**. The massive reactor pressure vessel (RPV) is an additional source of stored heat. These cannot be easily simulated in scaled experimental facilities typically made of relatively thin–wall vessels, and no such attempt was made. One should remark, however, that the importance of both heat release from the RPV and of heat "soaking" into the containment structures decreases with time (as the structures come into temperature equilibrium with their surroundings and exchange less heat). Thus, for the long–term behavior of the experimental facilities considered here, heat exchanges with the RPV and the containment structures do not constitute the dominant heat sink.

More specifially, for the DW, in the long term, heat exchanges with structures are not important except to the extent that they may produce temperature gradients from one place to another, which may influence mixing. The amount of heat transferred to DW structures would be totally negligible compared to removal via the PCCS. In the SC, the situation is different; the outer containment wall can become a significant heat sink for energy that comes by heat conduction and bypass leakage from the DW.

Fitch (1993) used the TRACG Code to estimate the heat release from the SBWR RPV and its contents to both the primary coolant and to the air space in the DW. The calculations show that the heat release is essentially complete one hour after the beginning of depressurization. The overwhelming fraction of the heat (some 40 full–power seconds) is released to the primary coolant, while a small fraction (roughly 0.7 full–power seconds) goes directly from the RPV outer wall to the DW.

In the case of GIST, the water inventory in the RPV was increased to compensate for RPV metal heat release and arrive at an adequate simulation of level swell in the vessel (Billig, 1989). Heat release from RPV structures was not explicitly modeled in GIRAFFE. However, in view of the fact that the GIRAFFE tests are related to system behavior at least one hour after LOCA, this can be considered to be a minor effect. The same could be said for PANDA, but the decision was made to include it, since the necessary calculations are available and it is easy to accommodate the metal heat release through a small adjustment in the programming of the RPV electric heaters.

The **heat losses** from the systems considered are a directly related issue. Because of its much smaller volume–to–surface ratio and larger heat capacity, heat losses from the SBWR are relatively much smaller than from the experimental facilities. It is important to measure accurately the heat losses in the experimental facilities for application of the test data to computer code qualification.

Heat losses in GIST were not measured but, on the basis of engineering judgment, supported by computer calculations, are believed to be of second-order compared to the dominant energy transfers which govern the behavior of the facility. In the early GIRAFFE tests, heat losses were significant but were carefully measured and were compensated by an increase in the electric heater power. In the second series of GIRAFFE tests, efforts were made to minimize the heat losses by instaling guard heaters below the insulation on certain system components (Vierow, 1993). The heat losses during the PANDA experiments are expected to be relatively low and will be defined on the basis of extensive measurements. Additional discussion regarding the heat losses for the various experiments can be found in Section 4.

The concerns regarding any influence (on the overall behavior of the system) of heat storage and release from the RPV and containment structures and/or of heat losses from the experimental facilities are of significance only if such influences distort system behavior and lead to states of the system in the experiments which differ significantly from the expected behavior of the prototype.

It is important to note that the heat capacity of both the SBWR containment and of the corresponding parts of the experimental facilities, and the effects of transient heat conduction in the various structures and/or losses from the system, can be considered in computer calculations with a system code. Since conduction calculations are very well understood and can be performed with the necessary degree of spatial detail, and the thermal resistance is dominated by insulation with known properties, no significant uncertainty is expected in such calculations.

The conduction calculations need the fluid-to-wall heat transfer coefficient (h.t.c.) as input. The value of this h.t.c. is important and may be limiting the soaking rate during the initial period of transient heat transfer to wall and structures, during which the heat flux can be high. Afterwards, the heat flux into the walls and structures is usually limited by conduction, rather than convective heat transfer to the surface. Thus, although there may be some uncertainty regarding the condensation h.t.c. in the containment volumes, this uncertainty will not affect the calculated heat soaking rate.

*In conclusion*, the structure heat storage and heat loss issues for the experimental facilities *can be addressed adequately via data reduction and by the system codes* for both the SBWR and the experimental facilities.

## 3.5 Scaling of the Vents

The main (LOCA) vents will operate during the blowdown phase of the accidents considered; this phase is investigated in the GIST tests. The dynamics of **main vent clearing** is not an issue for GIST since vent flow is well established by the time the RPV pressure falls below the initial pressure of these tests. The main vents are not normally expected to open during the long–term containment cooling phase.

The PCCS vents will inject mixtures of steam and noncondensables into the SP starting with the blowdown phase and continuing thereafter.

The important phenomena that must be considered to understand the operation and consequently to properly scale the vents are:

(a) Flow regime and formation of bubbles at the vent.

(b) Creation of single– or two–phase plumes from the vents.

(c) Entrainment and mixing of fluid from the pool into the rising plume.

(d) Residence time of the two–phase plumes in the pool.

(e) Condensation rate of bubbles or jets containing noncondensables.

(f) Average temperature of the fluid in the plume as it reaches the surface of the pool.

Items c and f were already considered in Section 3.2. The remaining points related to the creation and behavior of two–phase plumes from vents are discussed in Section 3.5.2.

### 3.5.1 Number of Vents, Flow Area and Vent Hydraulic Diameter

The scaling of the number of vents and vent dimensions (up to the location of submergence in the SC pool) is covered by the general scaling criteria for the piping (Section 2.4.3). The geometrical configuration of the vents and their dimensions *at the submergence point* can clearly play an important role. Two limiting cases can be considered:

If the vent acts essentially like a point source of mass and energy in the pool, the analysis presented in Section 3.2.1 and in the following sections shows that it is practically not possible to design a scaled vent reproducing the behavior of the SBWR. On the contrary, if the actual vents are designed with multiple orifices or as "linear" sources of injected mass and heat, then their scaling is straigthforward: a fraction of the vent corresponding to the system scale R should be included in the experimental facilities (Section 3.5.2).

### 3.5.2 Condensation of Steam and Noncondensable Mixtures Injected from Vents into the Suppression Pool

The effects of vent design and scaling on pool stratification were discussed in Section 3.2.1. Moody (1986) provides information useful for the scaling of discharges from vents.

The start of GDCS injection essentially cuts off **main (LOCA) vent flow**. Regarding possible injection of steam and noncondensables from the main (LOCA) vents into the SP during the post–LOCA transient, submergence is the most important parameter and is conserved in all the tests.

Coddington (1993b) has reviewed the **PCCS venting phenomena**. This section summarizes the findings.

#### 3.5.2.1 Creation of Bubbles at the Vents

The Laplace constant

$$\left(\frac{\sigma}{g(\rho_L - \rho_G)}\right)^{1/2}$$

determines the relative effects of buoyancy and surface tension $\sigma$ in relation to bubble formation at an orifice. The experimental evidence (e.g. Wallis, 1969, pp. 244–247) shows that, for orifice diameters larger than the Laplace constant, the diameter of the orifice does not control the size of the bubbles produced. For saturated water at 0.2–0.5 MPa, the Laplace constant is of the order of 2.4 to 2.3 mm. In practice, vent orifice diameters are going to be larger than these values. For larger orifices[e], the bubble volume is given by

$$V_b = 1.138 \, \frac{\dot{J}_G^{6/5}}{g^{3/5}}$$

where $\dot{J}_G$ is the volumetric flow rate of the gas through the orifice (Davidson and Harrison, 1963). The bubbles created at the orifice will likely be large enough to breakup into a swarm of smaller bubbles. Condensation of the steam should help break up the bubbles exiting from the orifices. Paul et al. (1985), in relation to pool scrubbing experiments, state that large bubbles will break up within 10 "globule"

[e] The diameter has to be smaller, however, than the minimum diameter for the so–called "total backflooding" or "weeping" regime (Bugg and Rowe, 1992). This limit is given by $D < \dot{J}_G^{2/5}/(4g)^{1/5}$. When backflooding occurs, the bubbles are created *inside* the pipe. This limit will not be relevant when multiple–hole arrangements are used.

diameters to smaller bubbles with a log-normal distribution and a mean diameter of approximately) 5.8 mm. Coddington (1993b) estimates that the break up length can be of the order of magnitude of the SBWR PCCS vent submergence (nominally 0.75 m). The distribution of the bubbles created by breakup will depend on the Morton number of the fluid (e.g., Clift et al., 1978)

$$Mo \equiv \frac{\sigma^3 \rho_L}{g \mu_L^4}$$

where $\mu_L$ is the viscosity of the liquid. Clearly, for prototypical fluids, the Morton numbers, and consequently the bubble size distribution, will be preserved in the test facility.

### 3.5.2.2 Scaling of Two-Phase Plumes in the Pool

The residence time in the SP of bubbles created at the PCCS vents and containing a mixture of steam and noncondensables determines their degree of condensation. This residence time depends not only on the rise velocity of bubbles in stagnant liquid (of the order of 0.25 m/s for the bubble sizes — after breakup — considered here), but also on the influence of the void fraction in the two-phase plume and on the plume rise velocity, which, in turn depends on the rate of entrainment of liquid in the plume.

Preserving bubble size and plume characteristic dimensions (i.e., the plume source strength), as well as plume height, will also result in prototypical entrainment rates into the plumes; the plumes will be perfectly similar.

### 3.5.2.3 Condensation Rate of Bubbles or Jets Containing Noncondensables

Coddington (1993b) reviewed the literature on the condensation rate of steam bubbles containing noncondensables. He finds that the complete condensation time for pure-steam bubbles varies from 10 ms for small bubbles at high pool subcoolings up to 1 s for large bubbles in a low-subcooling pool. The addition of noncondensables increases the condensation time by up to a factor of 2. For small bubbles (2-mm diameter), the complete condensation time is much shorter than their residence time in the pool. For large bubbles, however (20 mm diameter), the bubble collapse time is comparable to its residence time. The residence time will, of course, be similar for essentially identical plumes.

#### 3.5.2.4 Geometry of the Vents

We conclude that the behavior of the mixture of steam and noncondensables injected into the SP from the PCCS vents depends very much on the manner in which the total volumetric flow is distributed at the end of the vents. Within the range of volumetric flow rates of interest for the SBWR vents, the phenomena depend uniquely on the volumetric flow rate *per vent orifice*. In view of the dependence of individual bubble and two–phase plume behavior on bubble diameter, proper scaling of the vents is only possible when bubble dimensions are identical. Thus, it is possible to correctly scale the bubble generation and breakup phenomena at the vents, as well as the subsequent chain of phenomena, including condensation of the bubbles in the pool, *only when the vent orifice diameters are prototypical*. This is possible, for example, when a "linear" "line source" like vent arrangement is used. In this case, the plume in the model will simply be a segment of the plume in the prototype.

### 3.5.3 Vent Clearing, Chugging and Oscillations in the PCCS Vents

The dynamics of **main (LOCA) vent** clearing affect the peak containment pressure only during the early phases of blowdown. The main vents are not expected to open during the post–LOCA period, as already noted in Section 3.5. During the long–term decay heat removal period, any uncovery of the main vents will be driven by relatively *slow* increases in DW pressure and will be properly scaled by the correct submergence depths of the main vents.

The condensation of steam and noncondensable mixtures injected from the **PCCS vents** into the SP may lead to cyclic condensation phenomena. The scaling of vent geometry and dimensions and their effects on such phenomena were examined in Section 3.5.2. Proper scaling should be guaranteed if the vents in the experiments are a segment of the actual ("line source" vents) used in the SBWR.

A cyclic discharge of noncondensables from the PCCS vent was observed in GIRAFFE. Noncondensables apparently accumulate in the vent and the condenser tubes, the performance of the condenser degrades, the pressure rises and depresses the vent water level, the vent clears and discharges the noncondensables to the SP. The volume of the vent lines is likely to influence the period of this phenomenon (which is of the order of 100 s). Although the volume of the vent lines cannot be exactly scaled, as discussed in Sections 2.4.1 and 2.4.5, the scaling of the PANDA PCCS vent lines approaches the system scale (1:14 as compared to 1:25).

## 3.6 Heat and Mass Transfer in the ICS and PCCS Condensers

### 3.6.1 Condensation Inside the Tubes

The detailed database for low–pressure condensation heat transfer in the presence of noncondensables inside the PCC (or the IC) tubes is provided by the MIT and UCB single–tube data. These data were used to develop the condensation heat transfer model used in TRACG (Vierow and Schrock, 1991). The GIRAFFE, PANTHERS, and PANDA data provide mostly integral verification of the adequacy of this database. The GIRAFFE data were used to qualify the TRACG model (Andersen et al., 1993b). A limited number of local tube heat flux measurements are also foreseen for the PANTHERS tests.

Table 3–2 summarizes the ranges of the variables covered in the University tests and the corresponding expected ranges in the SBWR. It is evident that the single–tube data, obtained in tubes having prototypical dimensions, cover the range of interest.

**Table 3–2**

**Parameter Range Comparison Between the Single–Tube Tests and the SBWR Conditions[e]**

| | SBWR | UCB–1[f] | UCB–2 | UCB–3 | MIT |
|---|---|---|---|---|---|
| Number of runs | | | | | |
| Pure steam | | 0 | 6 | 30 | 0 |
| Steam/Air | | 30 | 30 | 50 | 52 |
| Steam/Helium | | 0 | 18 | 20 | 22 |
| Inlet pressure [psia] | 40–70 | 4–65 | 13–44 | 16–75 | 16–70 |
| Inlet temperature [°C] | 120–150 | 72–146 | 95–134 | 95–150 | 100–140 |
| Inlet steam flow rate [kg/hr] | 6–40 | 6–25 | 16–73 | 30–60 | 10–33 |
| Inlet air mass fraction | 0–0.4 | 0–0.14 | 0–0.4 | 0–0.4 | 0.10–0.35 |
| Tube dimensions | | | | | |
| Length [m] | 1.8 | 2.1 | 2.44 | 2.44 | 2.54 |
| Outside diameter [mm] | 50.8 | 25.4 | 50.8 | 50.8 | 50.8 |
| Tube thickness [mm] | 1.65 | 1.70 | 0.71 | 0.71 | 2.40 |

[e] from Schrock (1992, 1993)

[f] basis for TRAC–G correlation

### 3.6.2 Heat Transfer on the Secondary Side

Heat transfer on the outside of the PCC (and IC) tubes (i.e., at the secondary side) is affected by the fluid and flow conditions in the pool. The single-tube UCB and MIT data were obtained with well-defined single-phase conditions on the secondary side.

In both the experimental facilities and the SBWR a (varying) number of tubes are used and the fluid and flow conditions in the IC pools need to be considered. In particular, the following should be considered:

- Effect of boiling on the secondary side.
- Effects of natural circulation in the pool.
- Effects of entrainment of fluid from the pool into the tube bundle.
- Flow patterns and void fraction distribution within the tube bundle.

Natural circulation within the IC pool is tested at full scale in the PANTHERS facility, which has a prototypical pool size. In the smaller-scale facilities (GIRAFFE and PANDA), natural circulation in the pool can only be approximated. Comparisons of single-tube (UCB and MIT), GIRAFFE, PANDA, and PANTHERS data will provide information regarding the importance of the natural circulation and bundle flow pattern effects.

Parametric calculations were also performed with a detailed analytical model of a condenser tube to clarify certain issues. This model treats a single tube as a counter-current flow heat exchanger with subcooled or boiling water flowing upwards on the outside and a mixture of steam and noncondensable gas flowing downwards inside (Meier, 1992, 1993). The model was used to assess the importance of the conditions on the secondary (pool) side on the overall heat transfer rate. The parametric studies (Meier, 1993) show that the overall performance of the condenser tube is not affected at all by the secondary-side mass flow rate (i.e., by natural circulation in the pool) when the pool water is saturated. This is the condition that will be encountered most during the tests. The secondary mass flow rate affects the overall heat transfer rate only as the subcooling in the pool (considered as the "inlet" temperature in the calculations) increases; the influence remains modest, as shown in Figure 3-2. Thus, no great influence of the effects of natural circulation in the pool is expected and scaling of this phenomenon is not a major issue for the test facilities.

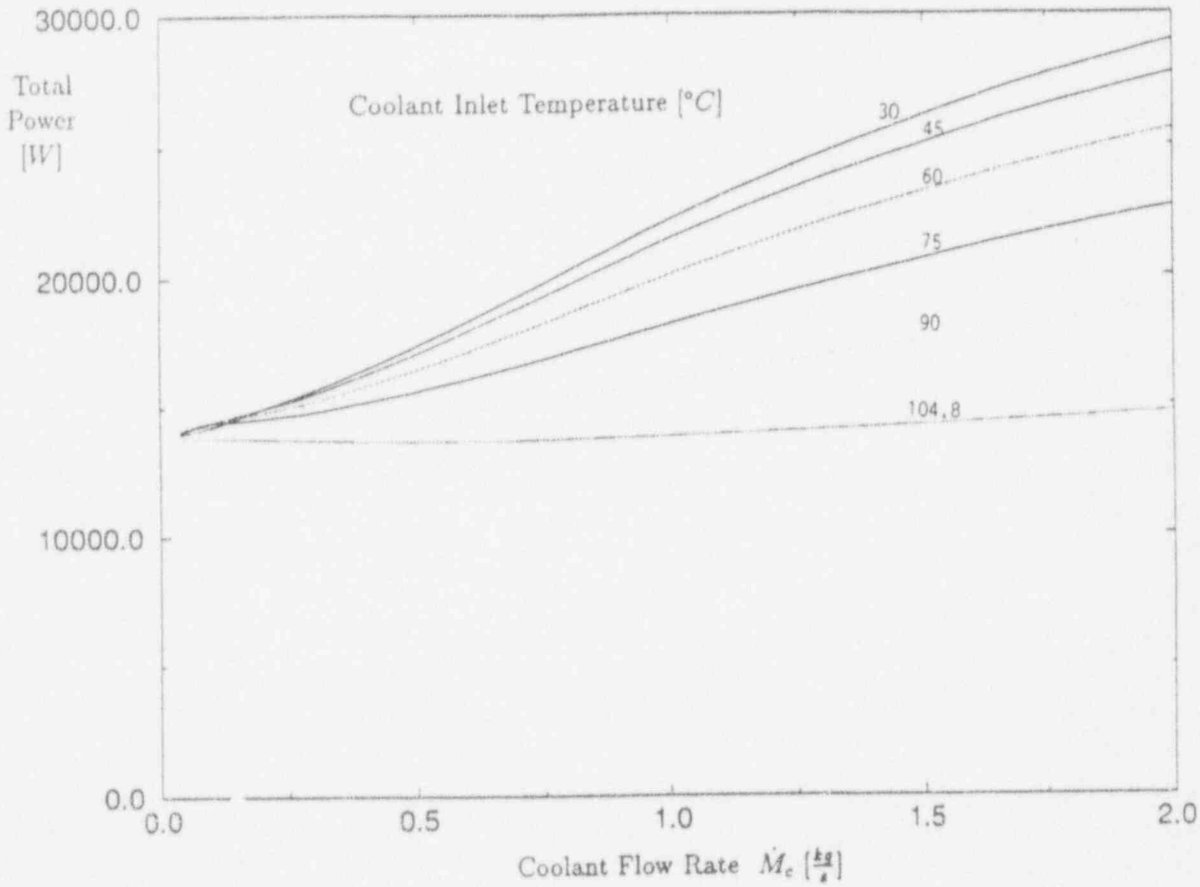



Figure 3-2 Effect of the Secondary Mass Flow Rate and Secondary Coolant "Inlet" Temperature on Condenser Tube Performance. [Calculations performed for the following typical conditions: pressure = 0.3 MPa; primary steam flow rate = 0.017 kg/s; nitrogen to steam ratio = 0.2; pool at 0.12 MPa.]

# 4. Scaling Approach for Specific Tests

The scaling approach followed in designing the various SBWR related facilities is briefly reviewed in this section, in relation to the main purpose of the tests. The use made of the data collected from these facilities to qualify the system code TRACG is discussed by Andersen et al. (1993a,b) and Kim et al. (1993).

## 4.1 GIST Tests

The purpose of the GDCS Integrated Systems Test (GIST) was to demonstrate the technical feasibility of the GDCS concept with scaled integral tests. The GDCS concept is based on the depressurization of the RPV to sufficiently low pressures to enable reflood of the core by gravity feed from an elevated pool. In particular, the tests were performed to show that the core remained covered under the most limiting design basis accident (DBA) conditions. The tests focused on system performance and on the coupled RPV–containment response for the low pressure (below 0.79 MPa) range of the LOCA blowdown phase.

More specifically, the GIST test objectives were to:

(1) Show the technical feasibility of the GDCS concept by performing a section-scaled integrated systems test of the SBWR design.

(2) Provide an additional database to qualify the TRACG code[a] for use in SBWR accident analyses.

(3) Provide data for refinement of the models available in the computer code TRACG to realistically predict the SBWR response to postulated LOCAs using GDCS injection for reactor makeup water.

When the GDCS operates, the gravity drain flow rate to the RPV depends on the piping geometry, the state of the fluid, and the pressure conditions in both the GDCS pool and the RPV. Flow entering the vessel during the later stages of blowdown during a postulated LOCA must be sufficient to keep the core flooded. The GDCS flow and core flooding phenomena *could* be studied in separate-effects tests if they were relatively decoupled. However, the degree of coupling could be better quantified with an integral test which included both flow from the GDCS pool

[a] The TRACG models that are relevant to GIST had already been qualified by other tests such as TLTA, FIST, etc (Andersen 1993a,b).

and flooding of a modeled vessel with a representative decay heat rate. Therefore, the GIST tests were designed to test the technical feasibility of the gravity drain process during and after system blowdown.

The GIST facility was built at the GE Nuclear Energy site in San Jose, California. All significant plant features which could affect the performance of the GDCS were included (Billig, 1989; Mross, 1989). Since the containment pressure and the GDCS pool water level determine GDCS activation, the containment (both upper and lower DW and SC volumes) was modeled. The facility is schematically shown in Figure 4–1. It is a 1:508 volumetric–scale, section model of the March 1987 SBWR conceptual design. Vertical elevations are scaled 1:1. Subsystem horizontal areas are also scaled according to the system scale of 1:508, except for the drain flow lines, as discussed in Section 4.1.2.1 below. Decay heat was modeled in proportion to the 1:508 system scale to provide the correct depletion of liquid water from the RPV by boiling (Billig, 1989).

The scaling of the tests produced data at real time and at prototypical pressures and temperatures. Detailed descriptions of the design and operation of GIST can be found in Mross (1989).

GIST has cylindrical vessels interconnected by piping simulating the various volumes of the SBWR, i.e. the RPV, upper DW, lower DW, and elevated SC (filling also the role of GDCS pool). The piping includes simulation of the Automatic Depressurization System (ADS), the GDCS lines, and the conditions at the break location for all break types considered (Billig, 1989).

The range of 24 tests conducted included Main Steam Line Break (MSLB), GDCS Line Break (GDLB), Vessel Bottom Drain Line Break (BDLB) and no–break (NB) transient tests.

During the GIST tests, the system was first depressurized to the atmosphere from its initial pressure of 1.07 MPa to 0.79 MPa. This period of the tests was thus used to create representative initial conditions in the RPV, as it entered the later stages of the depressurization transient (Figure 4–2). The initial conditions of importance are the decay heat generation rate, and representative water levels and void fraction distributions in the RPV. When the RPV reached the pressure level of 0.79 MPa, the blowdown flow rate was switched from the atmosphere to the DW for the break flow line, and to the SC for the ADS lines. Care was provided to obtain a smooth transition by not introducing changes in the blowdown flow areas.

With further depressurization of the RPV, the low pressure DPVs open. Eventually, the head of water in the SP becomes sufficient to overcome the RPV pressure and open the GDCS check valves allowing GDCS flow to enter and reflood the vessel.

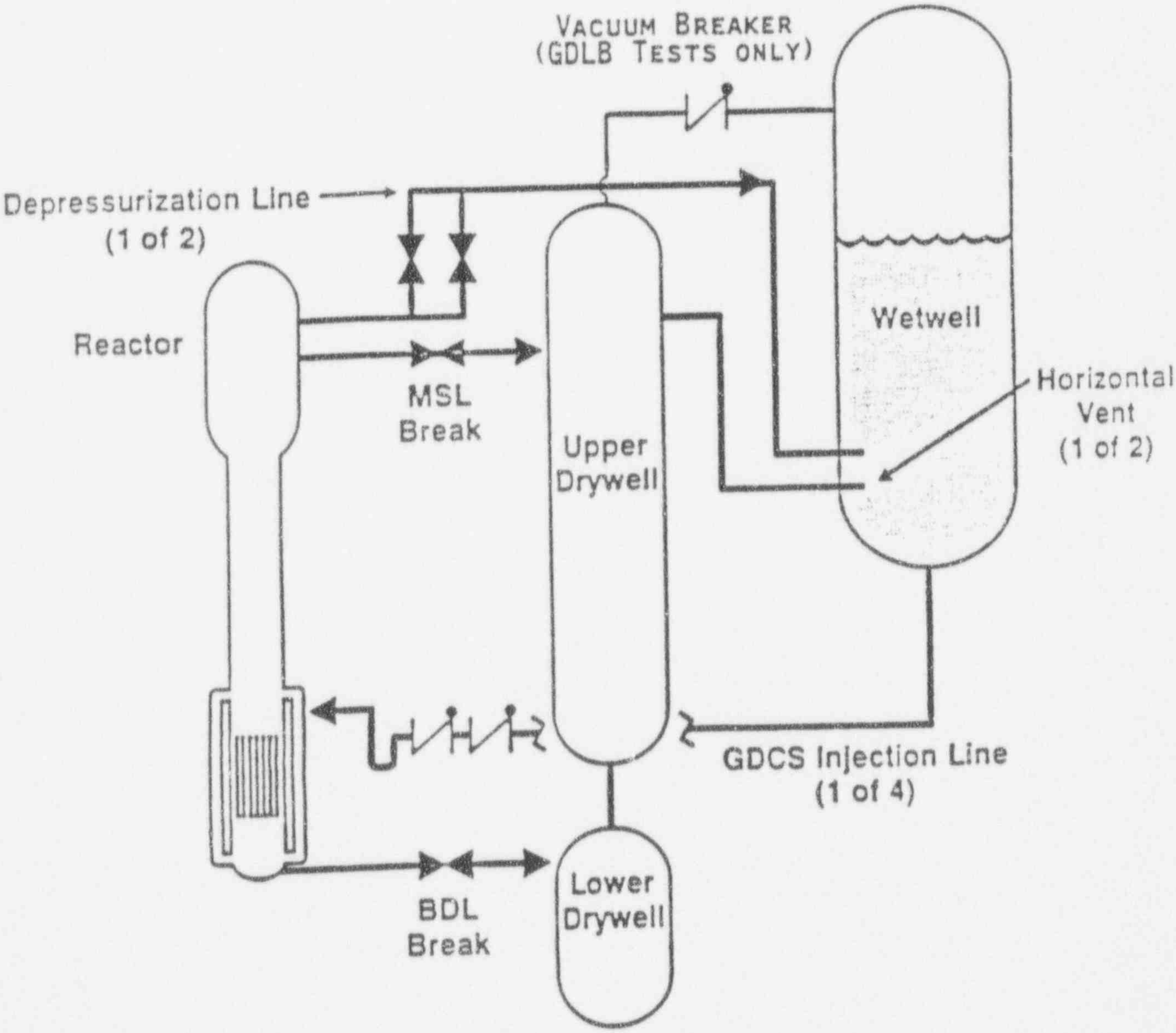



Figure 4-1 Main Components of the GIST Facility

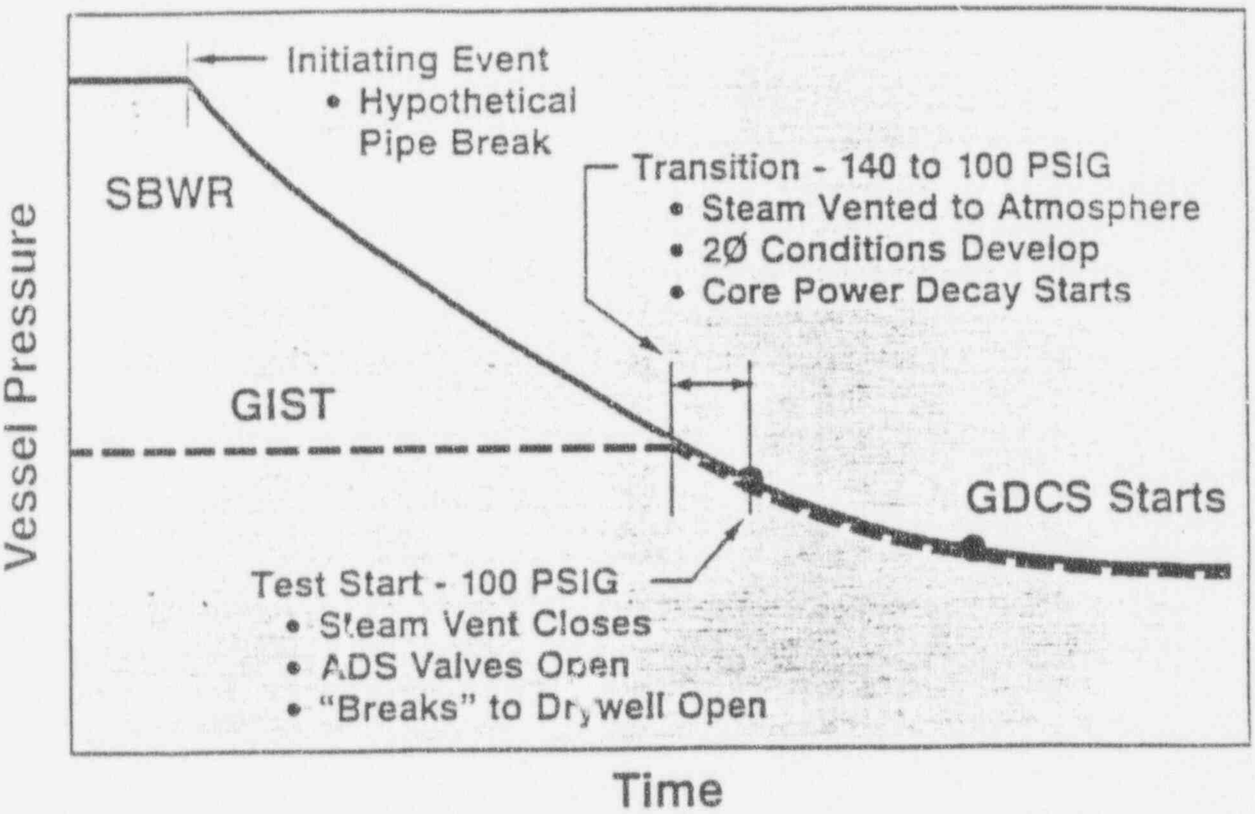



Figure 4–2 Vessel Pressure Coastdown During the SBWR and the GIST Depressurization Transients

### 4.1.1 General Scaling Approach

The design of the experimental facility is in agreement with the general top–down scaling criteria derived in Section 2. The bottom–up scaling of certain phenomena identified by the SBWR PIRT (Table 3.1) is discussed in Section 4.1.2 below.

The feasibility of the GDCS system would be demonstrated if it delivered sufficient core flooding flow during various LOCAs. The overall "global" system effects of pressure and RPV water inventory history during the LOCA had clearly to be modeled in a top-down fashion to include representative coupling effects of the GDCS flow with the RPV inventory. The GIST tests provided representative data that were used to qualify the TRACG code (Andersen et al., 1993). The particular phenomena not scaled in detail are not expected to produce bifurcations in system behavior invalidating the representativeness of the tests, or bringing the experimental apparatus outside the range of conditions expected in the SBWR; the differences are, furthermore, captured adequately in code calculations (Billig, 1989).

Billig (1989) discusses the limitations and scaling of the GIST tests; the Appendix of the report describes all the differences between the design tested and the final SBWR design, their impact on system performance, and justifies the validity of the GIST tests for SBWR applications.

As noted earlier, LOCA blowdown pressure history was simulated from a RPV pressure of 0.79 MPa, since the GDCS begins to function at lower pressure. The pressure-time characteristics were controlled by adjusting the flow area in the blowdown pipe for the various accident scenarios simulated.

### 4.1.2 Particular Scaling Issues for the GIST Tests

#### 4.1.2.1 Exact Scaling of the SBWR Geometry

There are a number of geometrical distortions in the GIST facility. These are due to the fact that the SBWR simulated design was not the final one; to the one-dimensional character of the facility and the relatively small horizontal-area scale of the system; and to the particular design choices of certain facility components. The experimental GIST results were used to qualify TRACG for SBWR analysis (Alamgir et al. 1990; Andersen et al., 1993b). Given the approach and limitations mentioned above, the GIST results should be viewed as having provided, in addition to a demonstration of the GDCS concept, additional data for code qualification. Most of the test limitations and particularities, as well as the code qualification work, were discussed already in detail by Billig (1989); this discussion is not repeated here. Only certain issues, such as the determination of the initial test conditions, are discussed below. Certain geometrical differences between the GIST facility and the final SBWR design are also discussed. Their impact was addressed by Billig (1989).

The major difference between the 1987 and final SBWR designs is that, in the earlier version, a single pool was used to provide both containment pressure suppression and the source of the GDCS water, Fig. 4-3. In the final SBWR design, the water inventories for GDCS and pressure suppression are separated. The suppression pool is in the SC and the GDCS inventory is equally divided among three pools located on the diaphragm floor within the DW. A second difference is the addition of six DPVs discharging directly to the DW, to supplement the ADS function performed by the SRVs discharging to the SP.

An important parameter determining reflooding of the vessel is the upward force acting on the gravity head of the coolant in the downcomer. This depends on the pressures in the RPV steam dome and at the GDCS flow injection point. The differences in the older design tend to *inhibit* GDCS injection in GIST, relative to what would be expected in the present SBWR. This can be understood by recognizing that the DW and SC pressures in the SBWR are *not* independent variables, Fig. 4-3. During the entire blowdown, the DW pressure exceeds the SC pressure by, at least, the hydrostatic head required to open the top LOCA vent (about 15 kPa). This means that the GDCS pool overpressure in GIST was actually lower, relative to the RPV pressure, than it would be in the plant. (Alternatively speaking, the pressure in the RPV steam dome was higher). Blowdown through the DPVs in the SBWR tends to sustain LOCA vent flow and ensures that DW pressure remains above SC pressure up to the time GDCS flow initiates. The top of the GDCS pool, which is directly connected to the DW will, therefore, be at a relatively higher pressure in the current SBWR.

The presence of the PCCS condensers in the SBWR will have a small effect on the *absolute* pressure in the DW, but will not change much the pressure *difference* acting to reflood the vessel. Similarly, the ICS condenser loop, absorbing some steam from the RPV dome, will tend to reduce the absolute pressure in the RPV but not the pressure difference causing reflood.

*In summary*, the combined effects of PCCS and ICS operation and of having separate GDCS pools in the SBWR DW may alter the absolute pressure in the RPV, but are expected to have minimal influence on the pressure difference driving the flow reflooding the vessel. The absolute pressure changes are small and are not expected to create atypical thermodynamic conditions. Finally, in judging the effect of differences between GIST and the final SBWR design, it is important to note that the key GIST parameters (i.e., RPV and GDCS water levels and containment pressure) were either representative or conservative relative to the final SBWR design. The key parameters were also varied over a sufficiently broad range to allow detailed qualification of the TRACG code.

The **GDCS drain line flow area** modeling was conducted according to the general top–down criteria of Section 2.4.5. The SBWR drain line flow areas were increased after the GIST facility was built. Since the drain flow rate depends on both the line area and flow resistance, it was possible to employ the original model pipes with orificing loss coefficients reduced (orifices removed). This resulted in a representative drain flow into the RPV, in line with the scaling described in Section 2.4.5: the

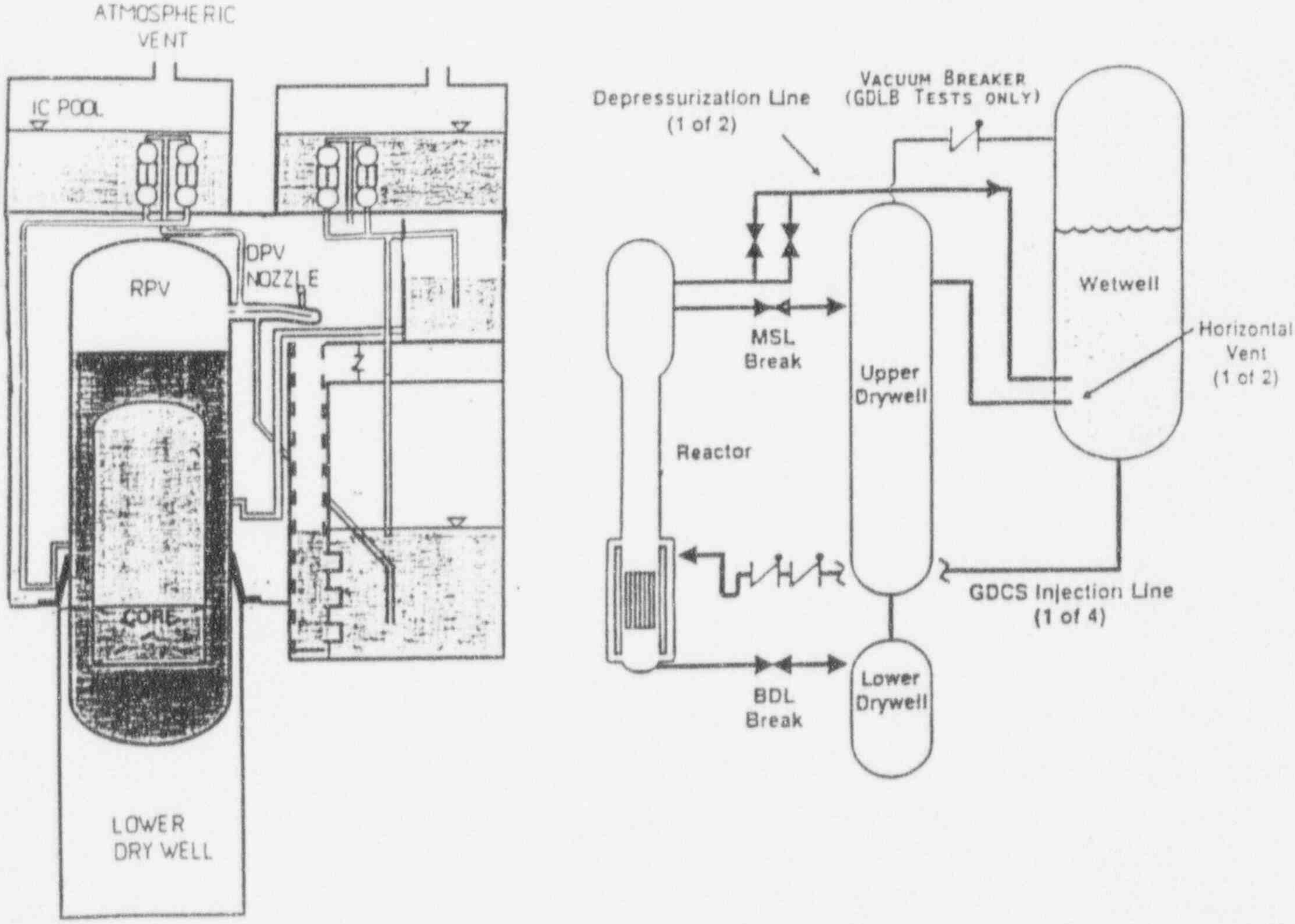



Figure 4–3 Flow Paths During the Late Blowdown Period in the SBWR and in the GIST Facility During a MSL Break

transit time of the fluid in the GDCS lines is short compared to the RPV filling time; as long as the flow rate remains representative, flow line modeling distortions related to transit time are acceptable.

**Heat losses** from the GIST facility were controlled by surface insulation. The losses were not measured, but, with the insulation used in the facility, they are estimated to be of secondary importance in relation to the major top-down-scaled energy transfers in GIST. For well insulated vessels, the estimation of the heat losses is relatively insensitive to uncertainties in the ambient convection heat transfer coefficient. The uncertainty in the heat loss estimation must be considered in code simulations of these tests.

#### 4.1.2.2 Establishment of Representative Initial Conditions

The initial test conditions for the GIST tests were determined from TRACG simulations of the early blowdown behavior of the SBWR from 7 MPa to 1.07 MPa. For the tests, the system was first depressurized to the atmosphere from this initial pressure of 1.07 MPa to 0.79 MPa. Thus, this period of the tests was used to create the representative initial conditions in the RPV as it entered the later stages of the depressurization transient, as shown in Figure 4-2. The pressure dropped from 1.07 to 0.79 MPa in 30 to 50 seconds (Billig, 1989); this provided sufficient time for representative conditions in the RPV to develop.

The vessels representing the containment were pressurized and preheated to the TRACG calculated pressures and temperatures. The DW was purged of air with steam to simulate the effect of air carryover into the SC.

Initial pool water temperatures in the GIST tests were controlled by heating prior to system blowdown. The initial temperatures ranged from 42 to 69 C, which embraces possible initial conditions in the SBWR.

The GDCS check valves, which open only when RPV pressure reaches a predetermined level, were not required to provide exact opening time characteristics, because the actual opening time is rapid relative to the vessel filling time.

The heat release from the RPV metal could not be simulated in the GIST tests; the heat stored initially in the RPV wall and its rate of release could not be scaled properly. Thus voids could not be maintained in the lower plenum and the water level in the core dropped; this was compensated by increasing the initial RPV water level in the tests. This distortion can be considered in TRACG calculations which can simulate the situation in the tests and in the SBWR.

## 4.2 GIRAFFE Tests

The GIRAFFE test facility (JAPC et al., 1990; Nagasaka et al., 1991; Yokobori et al., 1991; Vierow, 1993) is a full-height, reduced volume, integral system test facility, built by Toshiba at its Kawasaki City, Japan site, to investigate various thermal-hydraulic aspects of the SBWR passive heat removal systems, to demonstrate that the PCCS satisfies its design criteria in support of design certification, and to provide data for TRACG code qualification.

The facility consists of five major components representing the SBWR primary containment and IC pools, the PCCS condenser, and the connecting piping. Separate vessels represent the RPV, the DW, SC, GDCS pools, and ICS pools, which house the PCCS condensers. A schematic of the facility is shown in Figure 4-4 and details of its isolation condenser are shown in Figure 4-5. The facility scales the SBWR at a volumetric scale of 1:400. The heights and vent submergences are scaled 1:1. Pressure and temperature levels and pressure drops are preserved.

The objective of the GIRAFFE test program was to provide separate effects and integral data for qualification of TRACG. The **separate effects tests** address the issues of steam condensation heat transfer rates from a steam-nitrogen mixture under steady-state conditions, and of venting of noncondensable gases via the passive containment heat exchangers (PCCS system) to the SP. The **integral tests** demonstrate the concept of the PCCS for decay heat removal and provide data for a variety of LOCA simulations, against which computer codes for post-LOCA containment analysis may be qualified. Details of the scaling of the GIRAFFE facility and of the instrumentation are provided in the references cited above.

Data from separate effects condensation tests were obtained at a pressure of 0.3 MPa, for steam flow rates of 0.01 to 0.04 kg/s and nitrogen partial pressures of 0.0 to 0.03 MPa. The initial conditions for the noncondensable venting and long-term integral tests corresponded to those at one hour from the initiation of a LOCA. For the venting study, the nitrogen vent line of the PCC unit was submerged at depths of 0.4, 0.65, and 0.90 m, thus covering the range of interest for the SBWR.

The main steam line break, GDCS line break, and bottom drain line break LOCAs were simulated during the long-term system response tests.

### 4.2.1 Scaling of the GIRAFFE Facility

The detailed description of the facility can be found in the report by Vierow (1993), which is the basis for the following discussion. To comply with top-down scaling criteria, the GIRAFFE facility has been constructed on a 1:1 height scale to the

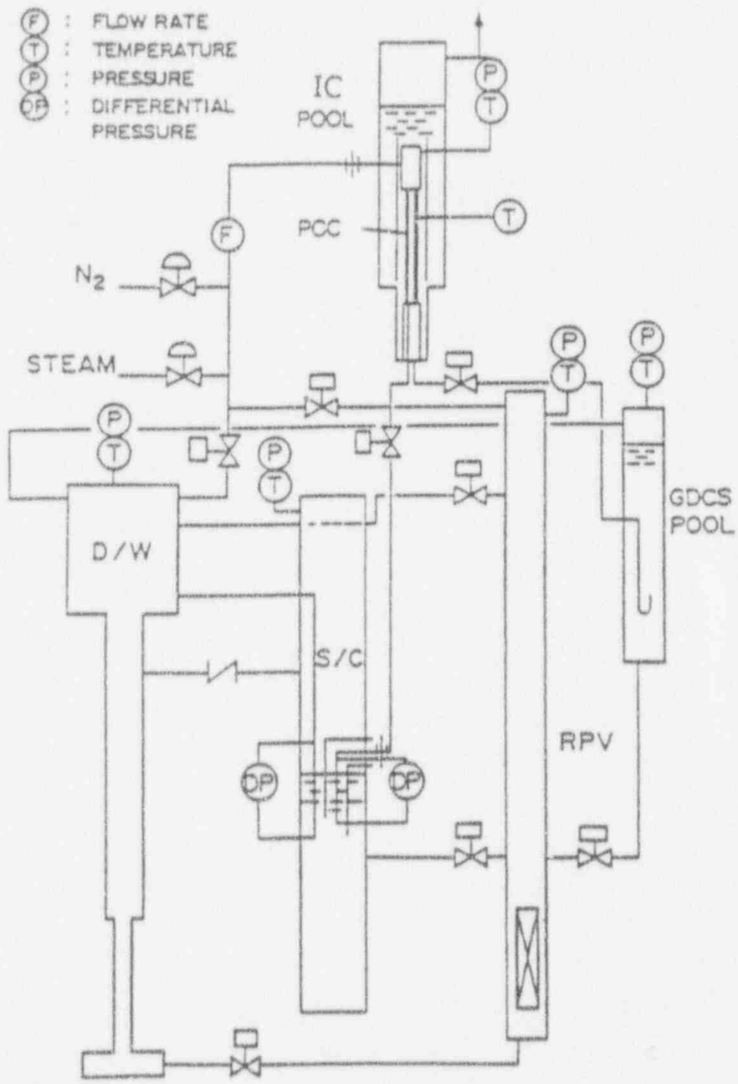



Figure 4–4 Schematic of the GIRAFFE Test Facility (Phase–2 Configuration)

SBWR design, and all essential elevation differences between the various vessels and between corresponding SBWR components have been preserved (Yokobori et al., 1991). The system scale for volumes, power, horizontal areas, and mass flow rates is 1:400. The RPV heater produced power according to a decay heat curve plus a constant amount, added to compensate the heat losses from the system (Vierow, 1993).

The RPV is simulated in full height from the bottom of the core up to the MSL exit, with the RPV–to–PCC and RPV–to–GDCS pool elevation differences conserved. The volumes above this full–height–simulation region have been shortened, but their volumes have been included in the volume of the RPV, which is scaled according to

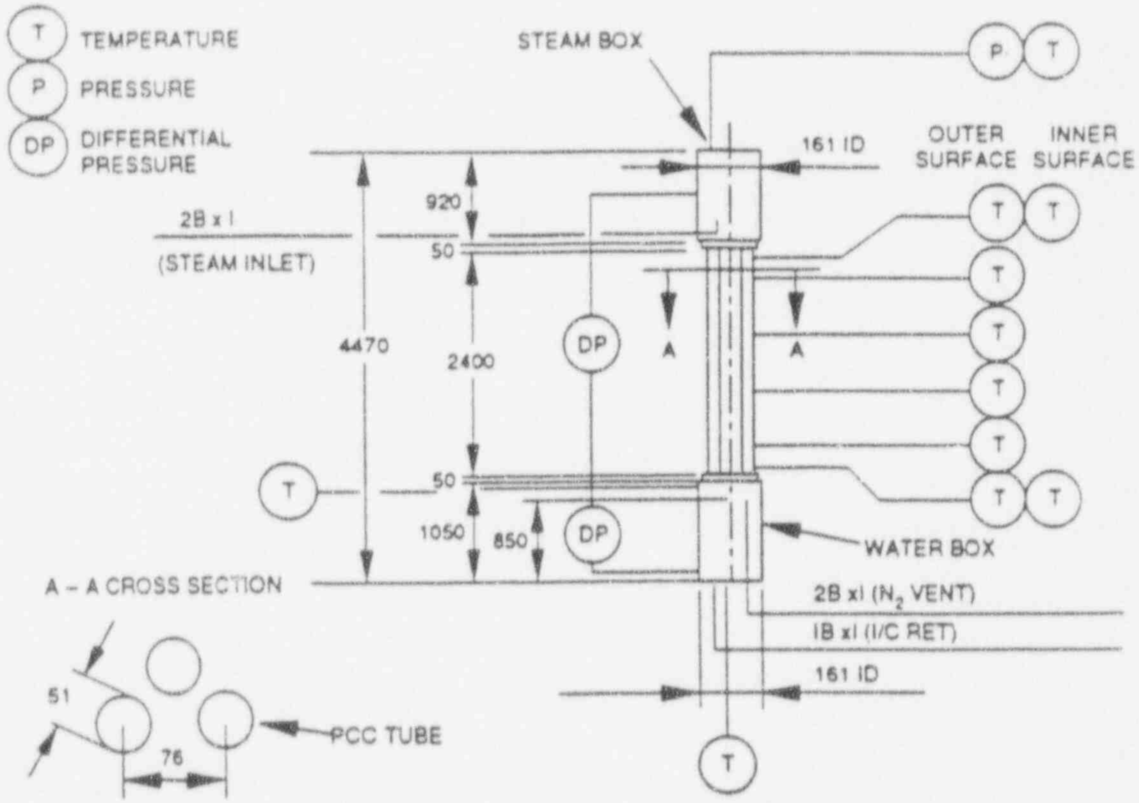



Figure 4–5 Details of GIRAFFE Isolation Condenser

the system scale of 1:400. The electrical heat input to the simulated core is also scaled 1:400.

The DW is also nearly full-height, with only the upper- and lower-most regions shortened. The volumes and areas are scaled down 1:400, and the cross-sectional area variation with height is preserved, so that the DW water level transient is similar to that expected in the SBWR under LOCA conditions. Although the annular shape of the SBWR DW could not be accommodated in GIRAFFE, the facility includes compartments that can be associated with the various regions of the SBWR drywell (Figure 4–4). The lower larger drywell volume, which represents on a 1:400 system scale the corresponding SBWR volume, can retain noncondensable gases and/or water. There is also a connection to a steam injection line, which is used to simulate evaporation from accumulated water. The narrowest region, representing the annular DW space around the RPV in the SBWR, has a correctly-scaled cross-sectional area; thus, water level changes in the DW occur at prototypical rates. A vacuum breaker line connects the annular DW to the SC gas space. Connections to the main steam line, the main LOCA vent line, the GDCS air space, the DPV line, and PCC steam supply line correspond to those in the prototype SBWR.

The full height of the SC has been preserved, while the gas and water volumes have been again scaled down 1:400. The LOCA vent line is at its actual SBWR elevation. The vertical section of the main LOCA vent is a close-ended pipe extending from the upper DW almost to the SC floor; each of the three holes in the pipe wall representing the main vents has the proper size and elevation. Three alternative PCC vents, each with a different submergence depth, were installed in the SP to allow the study of submergence effects. These vents are vertical pipes with open ends, installed near the vessel wall.

The GIRAFFE vacuum breaker connects the annular region of the DW to the gas space of the SC, as shown in Figure 4–4. This particular connection must be considered in code simulations of the system.

The GDCS pool has exactly scaled height and volume.

The GIRAFFE isolation condenser is a scaled representation of the three SBWR PCCS condensers; the GIRAFFE condenser has three tubes (Figure 4–5). This unit is somewhat different from the most recent PCCS condenser design, with a longer condenser tube length (2.4 vs 1.8 m) and vertical, cylinder-shaped inlet steam and condensate collector boxes. The tube wall thickness is 2.5 mm, compared to the 1.65 mm wall thickness of the SBWR PCC units. The total surface area of the condenser tubes in GIRAFFE is scaled by 1:372 or 1:386, based on the outside or inside tube diameters, respectively. This is sufficiently close to the system scale of 1:400, considering the fact that 1.8 m is the minimum length of the SBWR PCCS tubes. The three GIRAFFE condenser tubes are spaced as to maintain correct secondary-side cross-sectional flow area.

Matching the condenser tube surface area with tubes of different length does not provide *a priori* proper scaling. For moderate differences in length, however, the shorter length is compensated by the redistribution of the total available flow to the tube array. This was confirmed by calculations with the detailed analytical model (Meier, 1992, 1993) of a condenser tube mentioned in Section 3.6.2. The calculations showed that three 2.4 m tubes perform in a practically identical fashion as four 1.8 m tubes having the same total heat transfer area; deviations in the total heat transfer rate remained within 2 % over a wide range of operating conditions.

The IC pools house the PCC unit, which is placed within a chimney separating the region where the water is in contact with the PCC tubes from the outer pool region; this provides for simulation of the circulation in the IC pool. As stated in Section 3.6.2, the total heat transfer rate from the condenser tubes is relatively insensitive to the circulation pattern on the secondary side. There is enough water in the IC pool for three days of decay heat removal.

The initial test series in GIRAFFE were run with the PCC draining liquid directly to the RPV, while in the later series the present SBWR configuration (draining via the GDCS pool) was implemented. Such differences have a minimal effect on GIRAFFE's ability to study the PCCS phenomena and, furthermore, the differences can be easily accounted for in code calculations.

All lines are sized and orificed as to allow for prototypical pressure drops at scaled mass flow rates; the pipes are somewhat oversized with respect to the system scale. This reduces the frictional pressure drops, and the total resistance of each line is adjusted by inserting an orifice plate representing the appropriate loss coefficient. The adequacy of this scaling was discussed in Section 2.4.5.

## 4.2.2 Particular Phenomena of Relevance to the GIRAFFE Tests

### 4.2.2.1 Heat Losses from the Experiment

The GIRAFFE vessels, piping and flanges were insulated. The heat losses were measured under a variety of ambient conditions and are known to within a few percent. The heat losses from the RPV and connecting lines were compensated by increasing the power to the RPV heating element (Vierow, 1993).

In later tests, microheaters were installed on the GDCS, DW and SC vertical walls and on the SC roof beneath the insulation. The power to these heaters was regulated by a microprocessor, so as to maintain thermocouple pairs on the inside and outside of vessel walls at the same temperature and stop the heat loss.

### 4.2.2.2 Storage of Noncondensable Gas in the Lower DW

Relatively colder nitrogen may "fall" and possibly accumulate in the lower DW during certain phases of the simulated transients. This nitrogen may be subsequently convected upward, to alter the composition of the mixture entering the PCCS condenser, and consequently its performance. Nitrogen returned from the SC to the DW by opening of the vacuum breakers may also sink to the lower DW. The distribution of the noncondensables in the various DW regions is also affected, however, by any heat released from the RPV in the annular space surrounding it and by heat transfer to the colder vent wall; these details were not simulated in GIRAFFE. Water spilling from the DPVs tends to mix the DW gases and creates a saturated pool, which promotes upwards motion of the nitrogen. This effect was observed in GIRAFFE testing (Vierow, 1993).

A number of tests were run at the GIRAFFE facility to investigate the distribution and mixing of nitrogen in the DW and collect data for qualifying the TRACG Code (Vierow, 1993). These tests include a case with steam supply to the lower DW simulating the evaporation of water accumulated there. Special procedures were followed to charge the DW with various spatial distributions of noncondensable gas. Some DW mixing occurred prior to the start of the tests; to capture such pre-test mixing, the injection process must also be modeled in the computer simulations (Vierow, 1993).

#### 4.2.2.3 Scaling of the PCCS vents

As noted earlier, the GIRAFFE PCCS vents are round open pipes located near the vessel wall. They are likely to create wall plumes. As discussed in Section 3.5, this does not provide accurate scaling (in relation to a straight-pipe prototypical full-scale vent) for the single- or two-phase plumes emerging from the vents and for mixing in the pool. The comparison should be made with the final design of the SBWR vents.

## 4.3 PANDA Tests

The PANDA test facility, which is nearing completion, will be used to conduct integral system tests, as part of the ALPHA program at the Paul Scherrer Institute (PSI) in Switzerland (Coddington et al., 1992). It will demonstrate PCCS performance on a larger scale than GIRAFFE. The facility has a full 1:1 vertical scale, and 1:25 "system" scale (volume, power, etc). It is primarily intended to examine system response during the long-term containment cooling period. The overall objectives of the PANDA tests are to demonstrate that:

(1) the containment long term cooling performance is similar in a larger scale system to that previously demonstrated with the GIRAFFE tests,

(2) any non-uniform spatial distributions in the DW or SC do not create significant adverse effects on the performance of the PCCS, and

(3) there are no adverse effects associated with multi-unit PCCS and ICS operation.

The tests will also extend the database available for computer code qualification.

The initial test series at PANDA will consist of two main steam line break (MSLB) tests. The first will be similar to the GIRAFFE MSL break with uniform DW conditions, while the second is planned in a manner maximizing the influence of DW asymmetries on the operation of the PCCS condensers.

Uniform and asymmetric DW conditions can be created in PANDA through the capability to vary the fraction of break/DPV flow which is injected into each of the interconnected DW vessels. The steam condensing capacity directly connected to each vessel (i.e., the number of condenser units) can also be varied. Finally, steam and/or noncondensables can be injected at various locations in the DW vessels to study mixing phenomena and to provide envelope information for the corresponding SBWR conditions.

### 4.3.1 Conceptual Design

Early during the conceptual design phase of the facility, it was recognized that it is neither possible nor desirable to preserve exact geometrical similarity between the SBWR containment volumes and the experimental facility (Coddington et al., 1992). On the other hand, multidimensional containment phenomena such as mixing of gases and natural circulation between compartments may depend on the particular geometry of the containment building. The general philosophy followed in designing the experimental facility was to allow such multidimensional effects to take place by dividing the main containment compartments in two and by providing a variety of well-controlled boundary conditions (e.g., inbalances) during the experiments, so that the various phenomena could be studied under well-established conditions, and a behavior envelope of the system established. Carefully conducted parametric experiments will also provide more valuable data for code qualification, rather than attempts to simulate geometrically, but to a necessarily limited degree, the rather complex reactor system. Boundary conditions and the behavior of the interconnections between containment volumes can be controlled to study various system scenarios and alternative accident paths.

Thus the RPV and the GDCS pools are represented each by a vessel. The DW and SC are represented both by two separate, interconnected vessels, (Figure 4–6). The RPV contains a 1.5 MW electrical heat source. There is a total of three PCCS condensers representing the corresponding three units in the SBWR and a single ICS condenser representing two of the three SBWR units. (The two ICS condenser units correspond to the 2 x 50 % design value of the cooling capacity; the third ICS condenser is an extra 50 % redundant unit). The condensers are connected to the two DW vessels, as shown in Figure 4–7. The fact that there are three PCC units and only two DW volumes will allow some degree of asymmetric behavior or create flows between the two DWs, even with equal flow areas from the RPV to the two DW volumes.

The details of the system and its scaling rationale are described by Huggenberger (1991a, 1991b, 1992, 1993a, 1993b), Coddington (1992), and Coddington et al. (1992). Figure 4–7 shows details of the piping interconnecting the various volumes. The facility is heavily instrumented with some 560 sensors for temperature, pressure, pressure difference, level or void fraction, mass flow rate, gas concentration (humidity or air content), electrical power, and conductivity (presence of phase) measurements (Dreier, 1992).

### 4.3.2 Scaling of the PANDA Facility

The scaling of the facility conforms to the top–down and bottom–up criteria developed in Sections 2 and 3 of this report. Full vertical heights and submergences are preserved to correctly represent the various gravity heads; volumes are represented at the system scale. The exceptions to these are noted below (Huggenberger, 1991a, 1993a). The experiments will be conducted under reactor pressure and temperature conditions which are prototypical for the phase of the accident under consideration.

#### 4.3.2.1 Volumes

Figure 4–8 shows the geometrical arrangement of PANDA in comparison to the SBWR and the relative elevations of the two systems. All the SBWR heights are represented except those below the Top of Active Fuel (TAF) in the core. The top of the PANDA RPV electric heaters is placed at the TAF location; the heaters are, however, shorter.

In the RPV, the liquid inventory above the Bottom of Active Fuel (BAF) is scaled according to the system scale of 1:25. The *liquid inventory below BAF in the RPV* was eliminated (Figure 4–8), since it remains saturated and essentially inactive during the post–LOCA phase of the accidents considered, and is not required for the correct simulation of gravity heads. The *liquid volume between mid–core and BAF* is included, however, in the scaled PANDA RPV volume by a small adjustment of the vessel diameter (Huggenberger, 1993a). The lowest SBWR line simulated in PANDA is the equalization line which enters the RPV at one meter above TAF. Thus eliminating and redistributing the water volume below mid–core and modifying the length of the heater elements will not influence significantly any natural circulation paths.

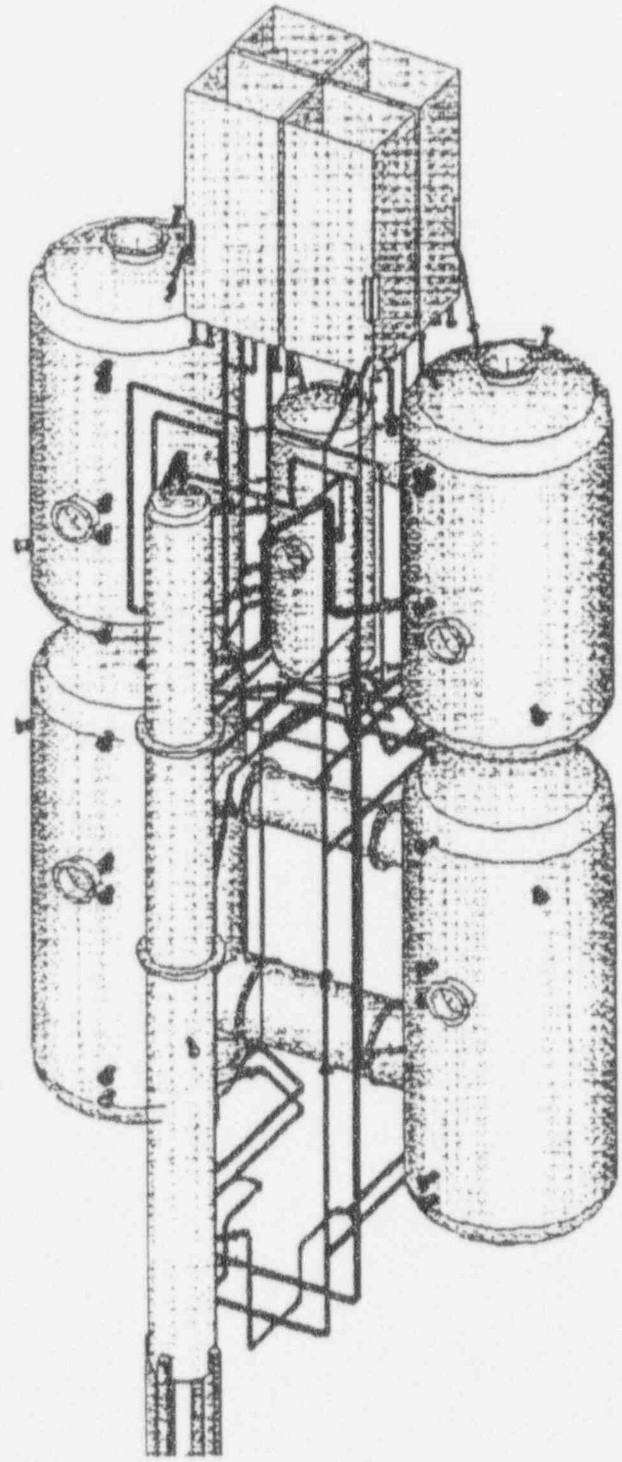


Figure 4–6 Isometric View of the PANDA Test Facility

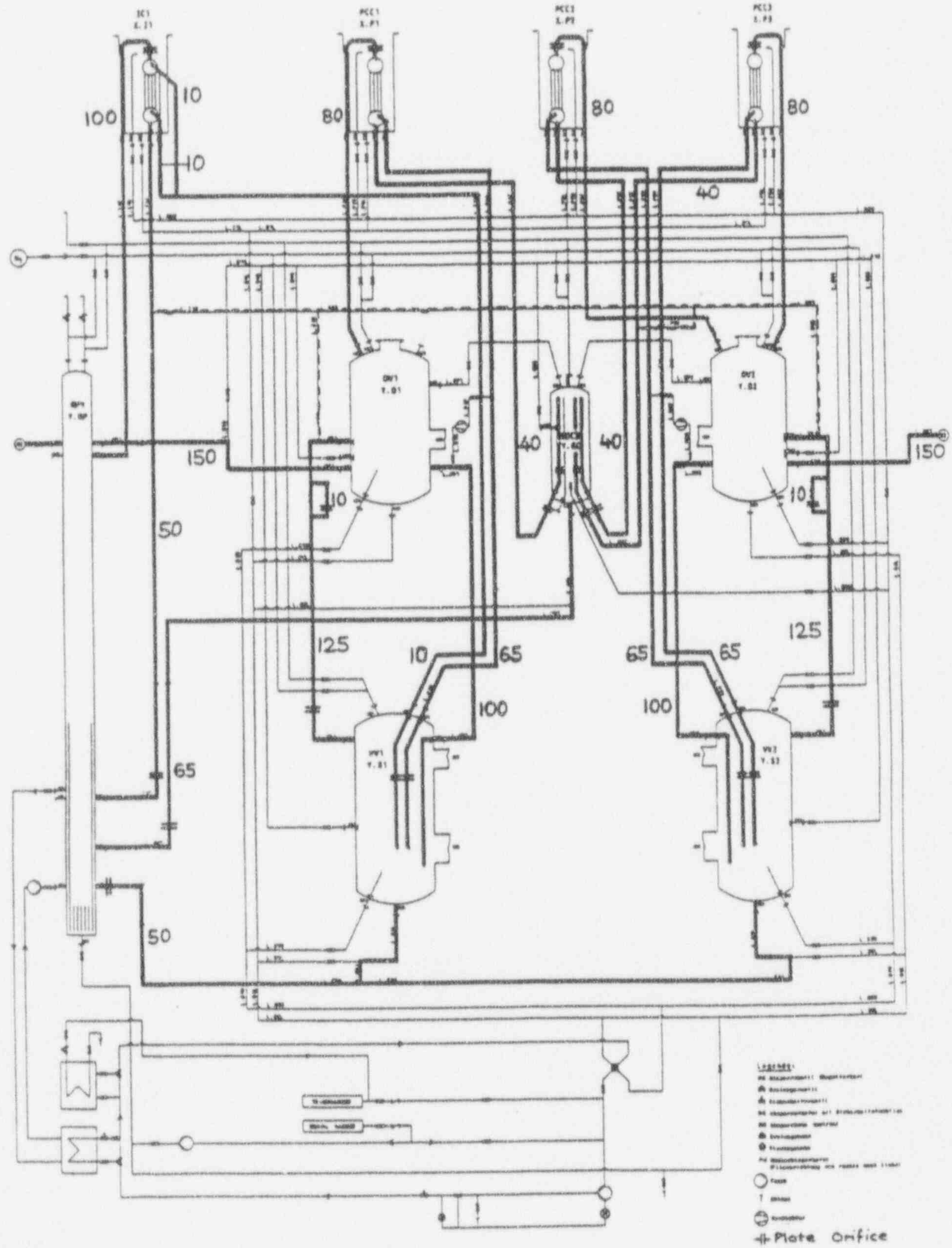



Figure 4-7 Piping Connections and Process Lines of the PANDA Facility

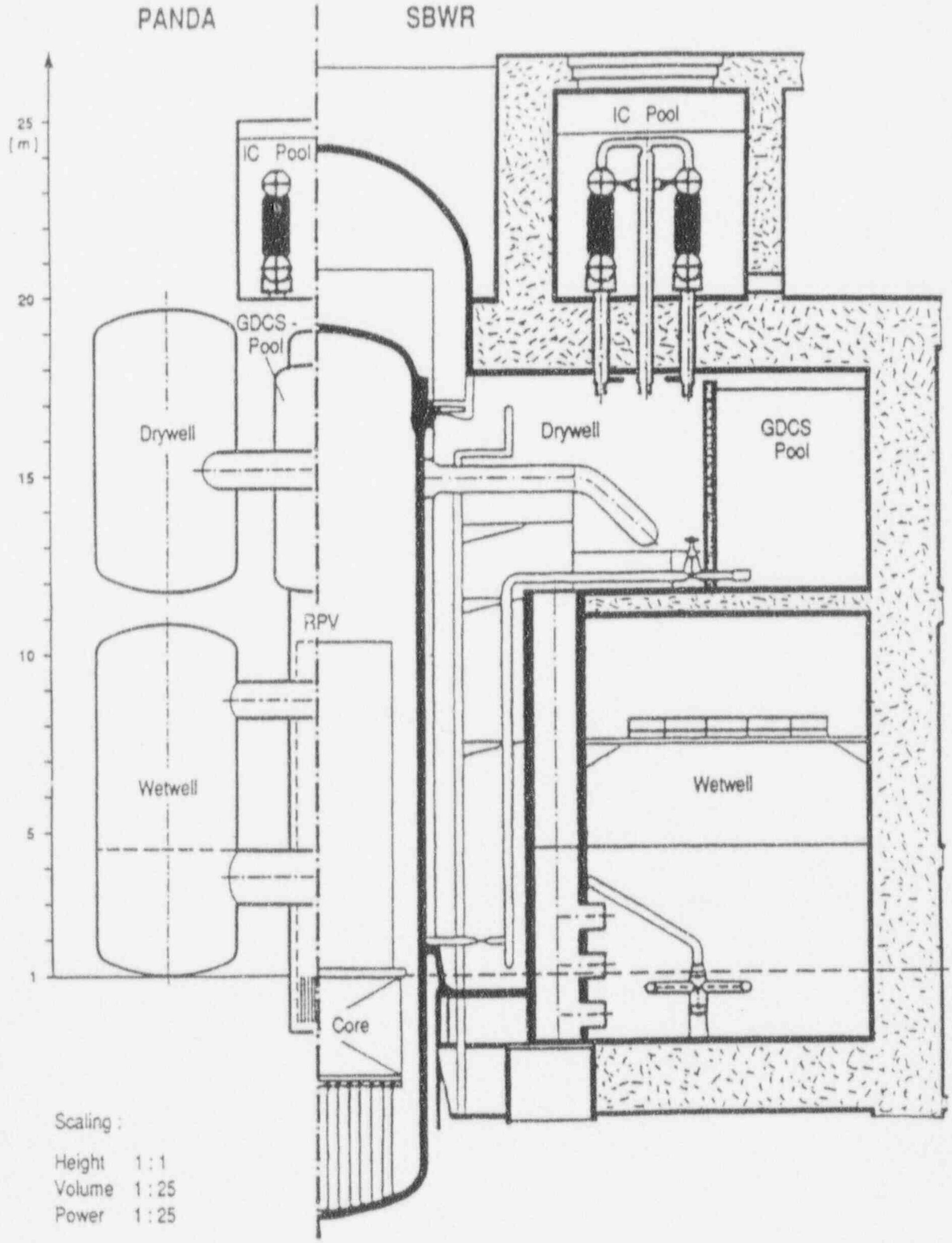



Figure 4-8 Comparison of the PANDA Elevations with the SBWR

The PANDA RPV includes a downcomer and a riser above the heater rods which represent the reactor core. The flow areas in the downcomer, the riser, and the core are scaled according to the top-down criteria of Section 2 (areas proportional to the system scale). The PANDA riser has no vertical partitions; its diameter is close to the hydraulic diameter of one partition of the SBWR riser. There is no representation of the steam separators and driers, because liquid entrainment and RPV to DW pressure drop are insignificant for the portion of the post-LOCA transients simulated by PANDA.

The *lower part of the water inventory in the SP* was eliminated to reduce vessel size. This water will not participate in the system thermal-hydraulic transient during the long-term cooling phase of the accidents considered. Indeed, the important phenomena will take place *above* the submergence depth of the PCCS vents. The PANDA PCCS vent lines are submerged in the SP with at least 2 m clearance above the bottom of the SC vessel, so that the reduced depth of this vessel will not influence venting of the noncondensables. Effects such as the convection of water to the bottom of the vessel by cold plumes running down the walls (Peterson et al., 1993) are of minor importance.

There is a depth of water of 2 m below the main vent submergence depth, that can accommodate any mixing below the vents. However, a *larger* fraction of the water inventory in the SP will absorb energy from the primary system *during blowdown*. This can be considered by properly defining the test initial conditions and preconditioning of the system accordingly.

The *lower part of the annular DW volume* surrounding the RPV was not included in the *height* of the PANDA DW volume, since it was felt that possible natural circulation phenomena taking place in this annular volume (heated on one side by the RPV) could not be adequately simulated. The volume of this annular space was, however, added to the PANDA DW *volume*. During the tests, air can also be injected at a controlled rate at the bottom of the PANDA DW to simulate the slow convection of nitrogen trapped in the annular part of the SBWR DW.

The *lowest part of the SBWR DW volume* (the region below the RPV support skirt and pedestal) is not included or added to the PANDA DW volume. Indeed, the lower DW volume provides only a "repository" for noncondensable gas or water. Its effect can be simulated in the same way as for the annular DW volume (controlled injection of air). The water inventory in the lowest part of the DW is significant only from the standpoint of producing long-term evaporation which could carry noncondensable gas to the upper DW and counteract the tendency of the noncondensables to sink to the bottom of the DW. In PANDA, this effect will be simulated by the combination of a saturated pool in the DW and controlled injections of air.

The GDCS compartment volume is smaller (1:64) than the system scale (1:25), since this volume does not play an important role in the dynamics of the system. In the transients considered, it simply provides a return path for the condensate to the RPV. The GDCS volume is about sufficiently large for containing the water inventory one hour after the LOCA. The horizontal surface area of the GDCS pool is also smaller (1:64) than the system scale. Thus while any tendency of the steam to condense on the surface of the GDCS pool water will be reduced, this will also lead to a slower heatup of the GDCS water. In terms of overall energy removal, the net effect should not be very significant.

Finally, the *volume of the ICS pools* is smaller than in the SBWR; these are scaled for 24 hrs of decay heat capacity, rather than 3 days as in the SBWR. However, water can be added at a required flow rate and temperature by the facility conditioning system to compensate for the lesser initial inventory.

#### 4.3.2.2 Scaled Models of the PCC and IC Condensers

A critical factor that led to the choice of 1:25 for the PANDA system scale was the requirement that the condenser unit secondary side behavior be representative of the prototype condensers. The PANDA condensers are "sliced" from the prototypes (Figure 4–9). Thus the circulation of the secondary coolant in a plane perpendicular to the axis of the cylindrical headers can be made very similar to that in the actual SBWR ICS pools. The units are provided with baffles, preventing entry of the flow into the bundle in the direction of the header axis. A sufficient number of tubes was provided to have at least a couple of tubes completely surrounded by other tubes. This led to five rows of tubes, twenty tubes in total, and to the 1:25 system scale. In PANDA, condenser tubes are in all respects (height, pitch, diameter, and wall thickness) prototypical (GE, 1992).

#### 4.3.2.3 Design of the Piping and Other Connections

All **piping** is scaled according to the criteria developed in Sections 2.3 and 2.4. The pipe diameters were calculated to match the frictional and form losses of the SBWR; the resulting pipe diameters were rounded to the next larger normalized diameter. The actual pressure drops are usually dominated by form losses which depend weakly on flow velocity (or Reynolds number) and can thus be matched very well (Huggenberger, 1992, 1993b). All lines are provided with interchangeable orifice plates that can be used to further adjust the pressure losses in the system.

The flow area of the PANDA main steam lines is the sum of the SBWR MSL's *and* the DPVs. Control valves are installed in each line (one to each DW) to simulate the different break geometries.

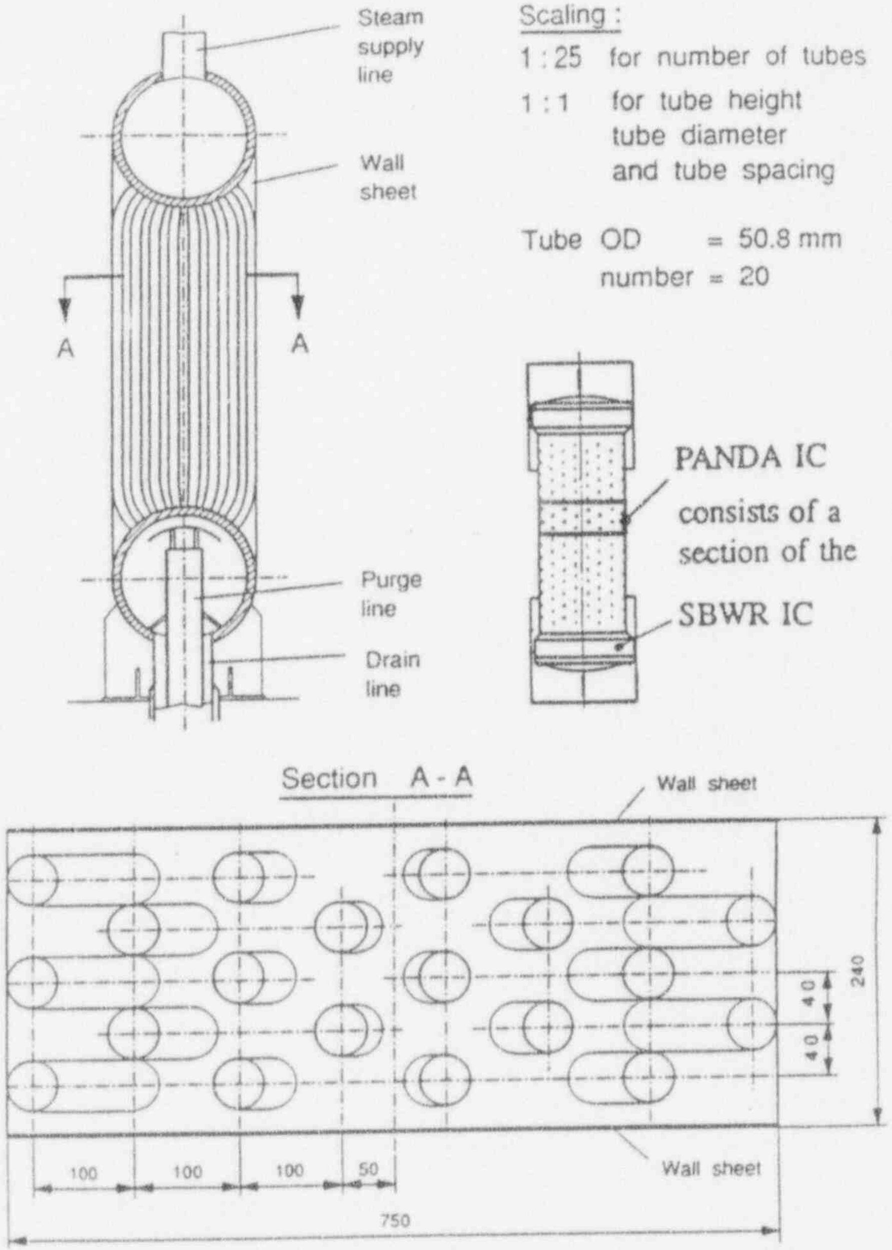



Figure 4–9 The PANDA PCCS and ICS Condenser Units

The **vacuum breakers**, which provide the flow path for potential redistribution of noncondensable gas between the SC and the DW, are simulated by programmable control valves which can reproduce the characteristics of the corresponding SBWR components. The scaling of the flows emerging from the vacuum breakers in relation to stratification in the DW was discussed in Section 3.2.3.

Finally, the **vents** are designed according to the criteria developed in Section 3.5. In PANDA, the vertical-pipe sections of the main vents have a cross-sectional area smaller than the one dictated by the system scale. However, they are not expected to open during the experiments. The gas velocities in the main vents are in both the prototype and the model low enough to eliminate worries about dynamic effects modifying system behavior, if the vents were to open.

#### 4.3.2.4 Heat Capacity and Heat Losses

The simulation of the heat capacity of the various SBWR structures was contemplated during the design phase of the PANDA facility. The PANDA vessels have thin walls and therefore very limited heat capacity. One could have inserted "heat capacity slabs" *in* the vessels to match the heat capacity of the SBWR structures. Use of appropriate layers of different materials *could* have provided the necessary time response. The idea was not implemented, however, since heat soaking in the SBWR structures during the long-term containment cooling period of interest is estimated to be of the same order of magnitude as the heat losses from the experiment. The heat capacity and heat loss aspects of the facility will be addressed by computation and use of calibration results during data reduction and analysis.

The PANDA vessels are very well insulated and the heat losses were conservatively estimated (for a 0.3 MPa saturated system) to be less than 4 % of the decay heat level one hour after shutdown and less than 9 % of the 24-hr-after-shutdown level (GE-PSI, 1991). More recent and more accurate estimates show that the actual losses will be significantly lower (Aubert, 1993).

The vessel internal and external wall temperatures are measured at 42 points (Dreier, 1993). This allows to determine accurately both the heat stored in the vessel walls and the heat losses. Heat loss calibration tests will be performed during commissioning of the facility. The information from these tests will be used to construct a heat-loss model of the facility. The data from the wall thermocouples will be used as inputs to the model to calculate the heat stored in the vessel walls and the heat losses during the tests; both are relatively small and both will be known with good accuracy. The heat-loss model of the facility will also be used for the code assessment analyses using the experimental data.

### 4.3.3 Establishment of the Proper Initial Conditions for the Tests

The PANDA facility will be equipped with auxiliary air and water supply systems for preconditioning the contents of the various system components; these are also shown in Figure 4-7. In particular, all vessels are provided with both top and bottom filling ports and drains or vents. Thus, the possibility of establishing stratified initial conditions in the water space of the vessels at the beginning of the tests is assured.

There is also the possibility of varying the submergence depth of the vents and of the initial water level in the SP. In particular, this water level can be positioned below, at, or above the location of the large pipe connecting the two SP in the SC vessels.

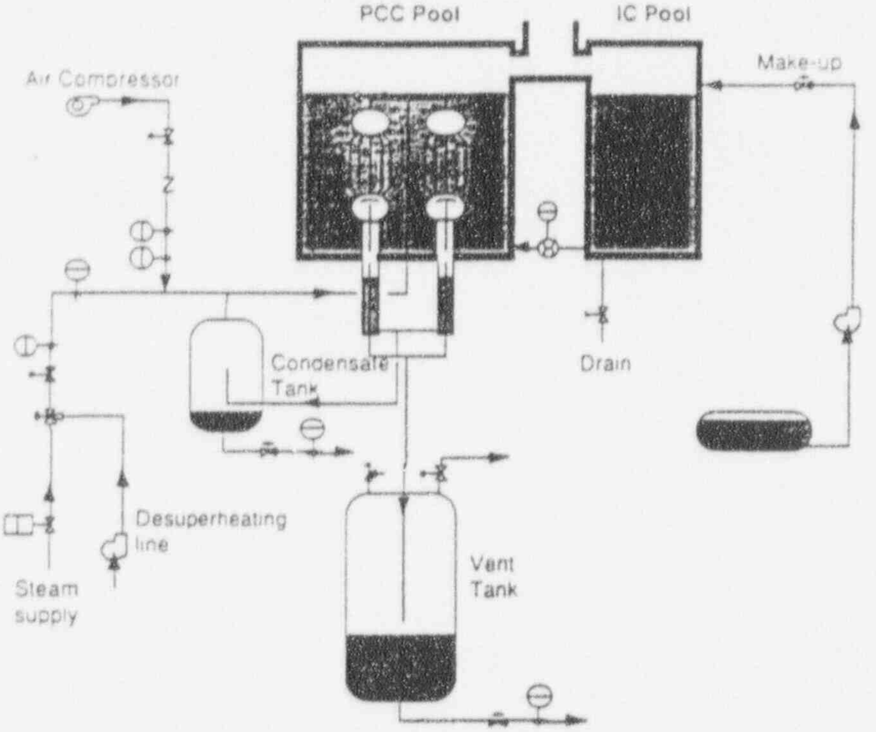
***In summary***, all the top-down and bottom-up scaling criteria derived in Sections 2 and 3 were applied properly in designing the PANDA facility. Any deviations from these (e.g., elimination of non-active fluid inventories) were discussed in this section. The PANDA model is essentially a 1:25 "vertical slice" of the SBWR. The heat capacity and heat losses of the experimental facility cannot be made to match those of the SBWR. This issue can be addressed, however, during data reduction by an accurate system heat balance based on measurements and heat loss calibrations.

## 4.4 PANTHERS Tests

PANTHERS is a full scale test under prototypical flow, pressure, temperature, and non-condensable-fraction conditions of a prototypical single module of an IC condenser (half unit) and of a full PCC condenser. The PANTHERS test facility and the planned tests are described by Masoni et al. (1993).

The purpose of the tests is to qualify the proposed condenser designs regarding structural integrity and steady-state thermal-hydraulic performance. Figure 4-10 shows the schematic of the PANTHERS test facility, as it will be configured for both types of tests.

The operation of the PCCS as part of the SBWR can be described as a slow transient. Under certain conditions, its operation may become cyclical, but the period of the cycles will be long in comparison to the response time of the PCCS. The characteristic response time of the PCC condenser unit is mainly determined by the transit time of the fluid in the tubes (which is of the order of seconds), and to some extent by the time constant of the tube wall; since these walls are thin, this time constant is also of the order of a few seconds. Thus the response of the PCC condenser units to changes in inlet conditions is much faster than the response of the large SBWR containment volumes. In terms of the discussion of Section 2.4.1



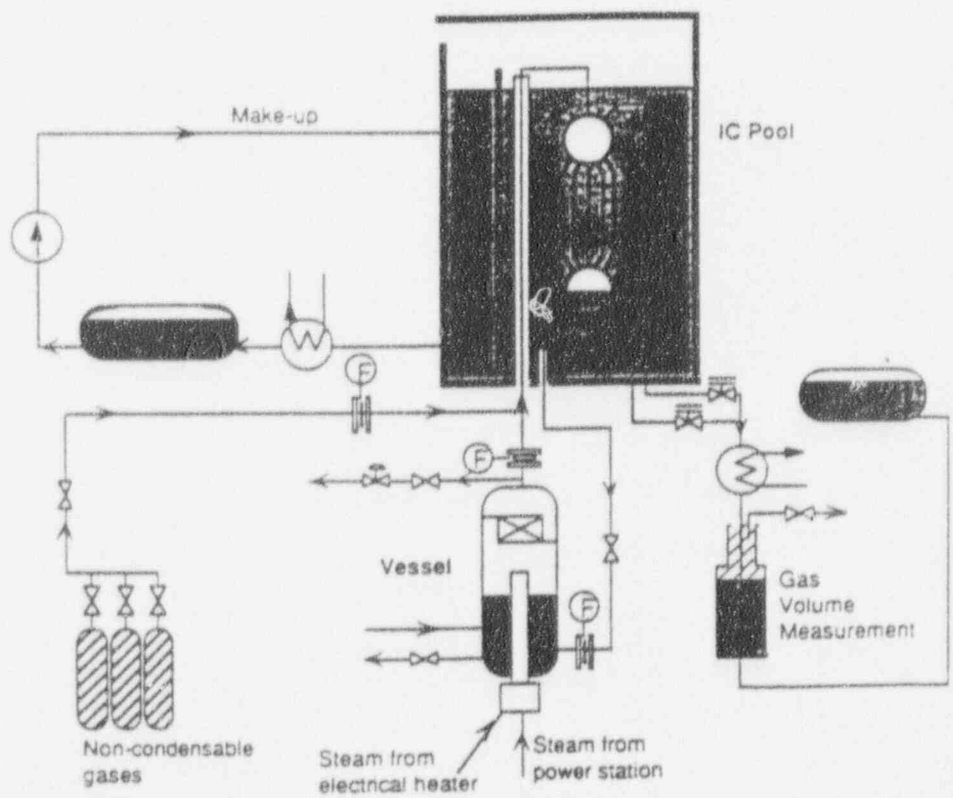




Figure 4-10 Schematics of the PANTHERS Test Facility [Top: PCC Tests; Bottom: IC Tests]

this can be expressed as $\tau^{o} \gg \tau^{o}_{tr,tubes}$. Thus the *steady state* PANTHERS tests provide adequate data to characterize the operation of the PCCS condenser units.

Some local heat flux data will also be obtained from these multi-tube units. Special care was used (Masoni et al., 1993) in installing thermocouples on both sides of the tube wall to reduce the relatively large error inherent in such measurements.

Since the PANTHERS tests are conducted with full-scale components (i.e., a prototypical PCC complete condenser unit and a symmetric one-half of an IC condenser unit), there are no scaling distortions to be addressed. There is no expected effect of testing only one half of the IC unit (one module), except possibly some influence on circulation in the pool.

The PCC condensers are installed in pools having the same dimensions as the SBWR IC pools. For the IC condenser tests, the pool surface is reduced by introducing a diaphragm wall to maintain the same area per module. Although this affects partly the boundary conditions regarding natural circulation in the pool (introduces a wall instead of a symmetry condition on one side of the unit), the effect should be minimal. In any case, small changes in the circulation patterns on the secondary side are not expected to have much influence on overall heat transfer, as discussed in Section 3.6.2.

The IC pools will be well mixed by natural circulation. The condensers are located at prototypical elevations in the pools. Furthermore, since the lower parts of the condenser units are located not much above the bottom of the pools, the entire pool water inventory will participate in the mixing process. Any unlikely stratification will be easily detected by the PANTHERS pool thermocouples.

Finally, note that any distortions due to testing only one module (half unit) will become apparent in the PCC condenser tests; for these, both a single module (half PCC unit) and the full PCC unit (two modules) will be tested.

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