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**MODULAR AND FULL SIZE SIMPLIFIED BOILING
WATER REACTOR DESIGN WITH FULLY PASSIVE
SAFETY SYSTEMS**

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Final Report

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U. S. Department of Energy Nuclear Energy Research Initiative

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**Final Report
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ACRONYMS

ABWR	Advanced Boiling Water Reactor
AC	Alternative Current
ADS	Automatic Depressurization System
BDL	Bottom Drain Line
BDLB	Bottom Drain Line Break
BNL	Brookhaven National Laboratory
BOP	Balance Of Plant
BWR	Boiling Water Reactor
CASMO-03	A neutronic code for lattice physics analysis
CCFL	Counter-Current FLOW
CHF	Critical Heat Flux
CPR	Critical Power Ratio
CPW-03	A neutronic code for lattice physics analysis
CRD	Control Rod Drive
DBA	Design Base Accident
DOE	Department Of Energy
DPV	De-Pressurization Valve
DW	Dry Well
GDCS	Gravity Drain Cooling System
GE	General Electric
GE-SBWR	600 MWe GE designed Simplified Boiling Water Reactor
HELIOS	A neutronic code for lattice physics analysis
HVBC	Hydraulic Vacuum Breaker Check
ICS	Isolation Condensation System
LOCA	Loss Of Coolant Accident
LWR	Light Water Reactor
MSIV	Main Steam Isolation Valve
MSL	Main Steam Line

MSLB	Main Steam Line Break
MVBC	Mechanical Vacuum Breaker Check
NERI	Nuclear Energy Research Initiative
NRC	Nuclear Regulatory Commission
NVG	Net Vapor Generation
PARCS	A neutronic code for nodal neutronic analysis
PCCS	Passive Containment Cooling System
PIRT	Phenomena Identification and Ranking Table
PUMA	Purdue University Multi-dimensional integral test Assembly
RAMONA-4B	A reactor safety analysis code
RELAP5	A reactor safety analysis code
RPV	Reactor Pressure Vessel
RWCU	Reactor Water Clean-Up unit
SBWR	Simplified Boiling Water Reactor
SBWR-1200	1200 MWe Simplified Boiling Water Reactor
SBWR-200	200 MWe Simplified Boiling Water Reactor
SBWR-600	600 MWe GE designed Simplified Boiling Water Reactor
SC	Suppression Chamber
SDC	Shut-Down Cooling system
SLCS	Standby Liquid Control System
SP	Suppression Pool
SRV	Safety Release Valve
SSAR	Standard Safety Analysis Report
TAF	Top of Active Fuel
TCV	Turbine Control Valve
TH	Thermal Hydraulic
TRAC	A reactor safety analysis code
TRAC_G	Modified TRAC code by GE
TRAC_M	Modified TRAC code

EXECUTIVE SUMMARY

The overall goal of this three-year research project was to develop a new scientific design of a compact modular 200 MWe and a full size 1200 MWe simplified boiling water reactors (SBWR). Specific objectives of this research were: (1) to perform scientific designs of the core neutronics and core thermal-hydraulics for a small capacity and full size simplified boiling water reactor, (2) to develop a passive safety system design, (3) improve and validate safety analysis code, (4) demonstrate experimentally and analytically all design functions of the safety systems for the design basis accidents (DBA) and (5) to develop the final scientific design of both SBWR systems, 200 MWe (SBWR-200) and 1200 MWe (SBWR-1200).

The SBWR combines the advantages of design simplicity and completely passive safety systems. These advantages fit well within the objectives of NERI and the Department of Energy's focus on the development of Generation III and IV nuclear power.

The 3-year research program was structured around seven tasks. Task 1 was to perform the preliminary thermal-hydraulic design. Task 2 was to perform the core neutronic design analysis. Task 3 was to perform a detailed scaling study and obtain corresponding PUMA conditions from an integral test. Task 4 was to perform integral tests and code evaluation for the DBA. Task 5 was to perform a safety analysis for the DBA. Task 6 was to perform a BWR stability analysis. Task 7 was to perform a final scientific design of the compact modular SBWR-200 and the full size SBWR-1200. A no cost extension for the third year was requested and the request was granted and all the project tasks were completed by April 2003.

The design activities in tasks 1, 2, and 3 were completed as planned. The existing thermal-hydraulic information, core physics, and fuel lattice information was collected on the existing design of the simplified boiling water reactor. The thermal-hydraulic design were developed. Based on a detailed integral system scaling analysis, design parameters were obtained and designs of the compact modular 200 MWe SBWR and the full size 1200 MWe SBWR were developed. These reactors are provided with passive safety systems. A new passive vacuum breaker check valve was designed to replace the mechanical vacuum beaker check valve. The new vacuum breaker check valve was based on a hydrostatic head, and was fail safe. The performance of this new valve was evaluated both by the thermal-hydraulic code RELAP5 and by the experiments in a scaled SBWR facility, PUMA.

In the core neutronic design a core depletion model was implemented to PARCS code. A lattice design for the SBWR fuel assemblies was performed. Design improvements were made to the neutronics/thermal-hydraulics models of SBWR-200 and SBWR-1200, and design analyses of these reactors were performed.

The design base accident analysis and evaluation of all the passive safety systems were completed as scheduled in tasks 4 and 5. Initial conditions for the small break loss of coolant accidents (LOCA) and large break LOCA using RELAP5 code were obtained. Small and large break LOCA tests were performed and the data was analyzed. An anticipated transient with scram was simulated using the RELAP5 code for SBWR-200. The transient considered was an accidental closure of the main steam isolation valve (MSIV), which was considered to be the most significant transient. The evaluation of the RELAP5 code against experimental data for SBWR-1200 was completed.

In task 6, the instability analysis for the three SBWR designs (SBWR-1200, SBWR-600 and SBWR-200) were simulated for start-up transients and the results were similar. Neither the geysering instability, nor the loop type instability was predicted by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillation was not observed at all because the power level used in the simulation was not high enough. A study was made of the potential instabilities by imposing an unrealistically high power ramp in a short time period, as suggested by GE. RAMONA-4B predicted core flow oscillations, of small amplitude, similar to that of the TRACG prediction by GE.

PROJECT OVERVIEW

The goal of this project was to develop two new scientific designs of next generation simplified boiling water reactors (SBWRs) namely, 1) compact modular 200 MWe SBWR and 2) full size 1200 MWe SBWR. The major objectives of this research were: (1) to perform scientific designs of the core neutronics and core thermal-hydraulics for a small capacity and full size simplified boiling water reactor, (2) to develop a passive safety system design, (3) improve and validate safety analysis codes, (4) demonstrate experimentally and analytically all design functions of the safety systems for the design basis accident (DBA) and (5) to develop the final scientific design of both SBWR systems, 200 MWe (SBWR-200) and 1200 MWe (SBWR-1200).

The means to achieve these objectives were integral tests and modeling simulations in seven major task areas: Task 1 was to perform the preliminary thermal-hydraulic design. Task 2 was to perform a core neutronic design analysis. Task 3 was to perform a detailed scaling study and obtain corresponding PUMA conditions for an integral test. Task 4 was to perform integral tests and code evaluation for DBA. Task 5 was to perform a safety analysis for DBA. Task 6 was to perform a BWR stability analysis. Task 7 was to perform the final scientific design of the compact modular SBWR-200 and the full size SBWR-1200. This final report covers the efforts and progress achieved during the this project period, August 1999 to April 2003.

The most important feature of the SBWR was the elimination of the re-circulation loop and pumps. The natural circulation cooling was provided in place of pumps. The natural circulation results in an extremely reliable and simple system to produce the steam needed to drive the turbine and generator. There are no active emergency core cooling systems. The reactor emergency core cooling systems are based on gravity-induced flow. Furthermore, the containment cooling was also performed by a passive system. Elimination of the re-circulation pumps and loops, internal pumps, and active safety systems substantially reduces the number of piping and valve components, and eliminates the need for a large emergency AC power supply as well. This simplification has considerable potential for reducing the cost of the reactor. In addition, the passive safety systems are more reliable providing enhanced safety against loss of coolant accidents and other design base accidents.

The SBWR has several advantages relative to the existing LWR systems. First, its safety systems are passive and eliminating the possibility of core uncover during design base accidents significantly reduces the probability of severe accidents. Second, the significant reduction in the number of pumps, and the elimination of the requirement for an emergency AC power supply, simplifies the plant design, operation and maintenance, as well as overall cost. Third, a full size SBWR can obtain an increased public acceptance in developed countries, whereas the compact modular SBWR, with fully passive safety features, was ideal for developing countries with a less developed industrial infrastructure.

The reactor safety systems in the SBWR are the gravity driven cooling system (GDCS) and the automatic depressurization system (ADS). The ADS was designed to rapidly depressurize the vessel following the receipt of a low vessel water level signal. This system was made up of both Safety Relief Valves and Depressurization Valves. The depressurization of the reactor vessel allows gravity injection by the GDCS. For long term cooling of the drywell (DW), an isolation condenser system (ICS) was adopted as a passive containment cooling system (PCCS). The steam from the DW was condensed through the PCCS condensers and was returned to the reactor vessel via the GDCS. The PCCS non-condensable vent line purges non-condensable gas into the suppression pool (SP). In the later stage of the blowdown phase during an accident, temperature stratification can be formed in the SP water. The PCCS purges the uncondensed steam to this hot temperature stratification layer. This may promote the temperature stratification and raise the SP pressure.

The unique feature of using two-phase natural circulation to remove the heat from the reactor core presents a complicated dynamic system. For example, the two-phase natural circulation rate can be significantly larger than the single-phase mode; however two-phase natural circulation can often be unstable, particularly at low pressure. Instabilities that can be encountered are the density wave oscillation, manometer oscillation, or a flashing induced cyclic flow instability. These instabilities can lead to flow conditions that are difficult to predict or control during transient or accident conditions. As the SBWR operates in a natural circulation mode, the occurrence of density wave instabilities should be avoided. The limit cycle and chaotic instability modes are not completely understood in the current and next generation BWR geometry. Hence in the design of the SBWR-200 and SBWR-1200 the reactor vessel natural circulation flow instability was studied in detail.

In the current design of the SBWR, the vacuum breaker check valve system was an important safety component. The failure of this mechanical component has a large bearing on the containment pressurization. A new passive vacuum breaker check valve, which was based on hydrostatic head, has been invented and developed to replace the mechanical vacuum breaker check valve.

The task areas from 1- 6 and have been covered during first, second and third years period. Within these defined tasks, thermal-hydraulic design, code modeling, integral testing, neutronic design, safety analysis efforts and investigation of start-up procedures were completed. This report presents efforts made, and progresses achieved. A no cost extension for the third year was requested and the request was granted and all the defined tasks were completed by April 2003.

1. INTRODUCTION

1.1 Significance of the Project

Nuclear power is now receiving wide spread acceptance as a clean and economic energy source in developed countries. Diversity is also needed to reduce reliance on hydro and fossil fuels. Hydro and fossil fuel power sources are limited due to geographical reasons, fuel supply and environmental concerns. Despite its abundance, nuclear power still suffers from a poor public image due to safety concerns and waste issues. The concern with the safety of nuclear reactors can now be reduced significantly by implementing advances in design, technology and operation. New passive safety systems have been proposed in the advanced reactor designs. By using all passive safety systems, a ten-fold increase in reactor safety is possible. Safer, simpler and intelligent engineering systems can be developed that are easier to operate and maintain. Furthermore, by using passive safety systems, the potential for serious human errors can be significantly reduced. These factors can lead to even more economical nuclear power systems.

There is a steady increase in the demand for electric power in developing countries. However due to the lack of engineering infrastructure in these countries, special attention should be paid to the application of nuclear power. It appears that reactors, which are very simple, passively safe, relatively small and modular in design and construction are preferred.

In view of this, the US Department of Energy (DOE) has been pursuing evolutionary and revolutionary design concepts for the future generation light water reactors. Key requirements in this pursuit have been a simpler and safer design, low development cost, rapid commercialization potential, and low construction and operation cost. The advanced boiling water reactor (ABWR) was the successful evolutionary design of the BWR. Currently two General Electric (GE) designed ABWRs are operating in Japan. The DOE has supported the revolutionary design in the AP600 pressurized water reactor. The Nuclear Regulatory Commission (NRC) has now completed the design certification for the AP600 by Westinghouse. A revolutionary design for the BWR is the Simplified Boiling Water Reactor (SBWR) developed by GE [1.1]. GE began development of a 600 MWe class SBWR with passive safety features, and was initially chosen by the DOE for design and certification as a standard plant. The SBWR was developed as a cooperative effort involving an international team of BWR vendors, architect-engineers, utilities,

universities and research organizations. The SBWR design has incorporated advances in proven technologies that have been developed over many years of commercial nuclear plant operation. The most important feature of the SBWR is the elimination of the re-circulation loop and pumps. Natural circulation cooling is provided in place of pumps. Natural circulation results in an extremely reliable and simple system to produce the steam needed to drive the turbine and generator. There are no active emergency core cooling systems. The reactor emergency core cooling systems are based on gravity-induced flow. Furthermore, the containment cooling is performed by a passive system. Elimination of the re-circulation pumps and loops, internal pumps, and active safety systems substantially reduces the number of piping and valve components and eliminates the need for a large emergency AC power supply. These simplifications have considerable potential for reducing the cost of the reactor. In addition, the passive safety systems are more reliable providing enhanced safety against loss of coolant accidents and other design base accidents. However, it appears that the 600 MWe size SBWR may not have a large market demand.

In view of the above, this research focused on two new designs, namely 1) a compact modular 200 MWe SBWR and 2) a full size 1200 MWe SBWR. These SBWR designs have several advantages relative to the existing LWR systems. First, the BWR is a direct Rankine cycle, which eliminates the need for steam generators. Second, a compact modular SBWR with fully passive safety features is ideal for developing countries. The compact reactor, because of its modular construction approach, requires a shorter construction time, is transportable, can fit remote site application, and requires fewer infrastructures. The full size reactor has a potential market in the developing or developed countries, including the Far East region. Passive safety systems enhance the safety level and hence wider public acceptance of the reactor. Third, the significant reduction in the number of pumps and elimination of the requirement for an emergency AC power supply simplifies the plant design, operation and maintenance, as well as overall cost.

1.2 Passive Safety Systems in SBWR

The SBWR uses passive safety systems. The reactor safety systems are the gravity driven cooling system (GDCS) and the automatic depressurization system (ADS). The ADS is designed to rapidly depressurize the vessel following the receipt of a low vessel water level signal. This

system is made up of both Safety Relief Valves and Depressurization Valves. The depressurization of the reactor vessel allows gravity injection from the GDSCS. For long term cooling of the drywell (DW), an isolation condenser system (ICS) has been adopted as a passive containment cooling system (PCCS). The steam from the DW is condensed through the PCCS condenser and is returned to the reactor vessel. The PCCS non-condensable vent line purges non-condensable gas into the suppression pool (SP). In the later stage of the blowdown phase during an accident, temperature stratification is formed in the SP water. PCCS purges the uncondensed steam to this hot temperature stratification layer, which then raises the SP pressure.

The unique feature of using two-phase natural circulation to remove the heat from the reactor core presents a complicated dynamic system. For example, the two-phase natural circulation rate can be significantly larger than the single-phase mode, and two-phase natural circulation can often be unstable, particularly at low pressure. Instabilities can be encountered such as the density wave oscillation, manometer oscillation, or a flashing induced cyclic flow instability. These instabilities can lead to flow conditions that are difficult to predict or control during transient or accident conditions. As the SBWR operates in a natural circulation mode, the occurrence of significant density wave instabilities should be avoided. The limit cycle and chaotic instability modes are not well understood. Hence in the design of the SBWR-200 and SBWR-1200 the reactor vessel natural circulation flow instability will be studied in detail.

In the current design of the SBWR, the vacuum breaker check valve is an important safety component. The failure of this semi-passive component has a large bearing on the containment pressurization. The present research provided an innovative design to replace this component using gravity based passive hydraulic valve system.

1.3 References

- 1.1 GE Nuclear Energy, "SBWR Standard Safety Analysis Report," Report No. 25A5113 Rev. A, August, 1992.

2. OBJECTIVES AND TASKS

2.1 Objectives

The main goal of this research project was the development of the scientific design of a compact modular 200 MWe and a full size 1200 MWe simplified boiling water reactors (SBWR). Specific objectives of this research were:

- (1) To develop preliminary scientific designs of the core neutronics and thermal-hydraulics for a small capacity and full size simplified boiling water reactor;
- (2) To develop a completely passive safety system design;
- (3) To improve and validate safety analysis codes;
- (4) To demonstrate experimentally and analytically all safety systems for design basis accident (DBA); and
- (5) To develop the final scientific design of both SBWR systems, SBWR-200 and SBWR-1200.

2.2 Tasks

Under these objectives the following seven tasks were defined.

- (1) Perform the preliminary thermal-hydraulic design for the compact modular SBWR-200 and full size SBWR-1200. Under this task the following subtasks are defined.
 - Perform design analysis for steady state coolant requirements.
 - Perform design analysis for safety requirements based on the design base accidents (DBA).
 - Define and design all passive safety systems.
- (2) Perform the core neutronic analysis and develop the design of the core through the physics calculations.
- (3) Perform a detailed scaling study and obtain corresponding PUMA conditions for integral tests.
- (4) Perform integral tests and code evaluation for DBA. Under this task the following subtasks are defined.
 - Demonstrate the function of safety systems.

- Develop a safety analysis database
 - Evaluate the safety system code capability.
 - Validate the code applicability and code improvements
- (5) Perform safety analysis for DBA.
- (6) Perform BWR stability analysis. Both steady state and start-up transients are studied in this task.
- (7) Perform the final conceptual scientific design of the compact modular SBWR-200 and the full size SBWR-1200.

3. PROJECT MILESTONES

3.1 First Year Milestones and Technical Tasks

Here, the milestones for the first phase of the project are presented. The first phase of the project was performed during the first year period. These milestones are listed below along with their respective technical tasks.

- Preliminary scientific design of the SBWR-200 and SBWR-1200: The technical tasks here are the development of the preliminary design of the thermal-hydraulic system and passive safety system.
- PUMA scaling study: Here the task is to perform detailed static and dynamic scaling of the thermal-hydraulic phenomena and system geometry using first principles.
- Detailed research and design concentrated on SBWR-200: Here the detailed scientific study of the SBWR-200 system components, governing thermal-hydraulic and neutronic behaviors and safety systems are analyzed.
- Large break and small break LOCA study: Technical task under this milestone are obtaining the initial conditions to conduct integral tests on PUMA facility using REALP5/MOD3 code, performing integral tests as per well established procedures and quality assurance program, and analyzing the data.
- Preliminary core neutronic design: Here the technical tasks involve obtaining the neutronic codes (HELIOS, PARCS) and benchmarking the codes against standard core design, and development of the SBWR-200 reactor core through calculation on assembly level.
- Preliminary study on BWR instability: Under this milestone the technical task is to analyze the SBWR-600 and SBWR-200 normal and abnormal start-up and instability using RAMONA-4B code.

3.2 Second Year Milestones and Technical Tasks

Here the milestones for the second year of the project are presented along with their respective technical tasks.

- SBWR Design Improvement: The objective of this task is to provide design improvements to the SBWR-200 and SBWR-1200. On the safety system improvement, a new design of

vacuum breaker check valve was developed. The new valve is based on hydraulic head unlike the previous design, which utilizes a mechanical valve in order to eliminate the potential of the malfunction of the vacuum breaker. The performance of the new valve was first evaluated using the RELAP5 code for the SBWR-200 system. Then the passive vacuum breaker check valve was constructed and implemented in the PUMA facility. It was then tested during the large break LOCA transient. The new design eliminated potential containment over-pressurization by the vacuum breaker malfunction.

- Detailed research and design concentrated on SBWR-1200: Here the detailed scientific study of the SBWR-1200 system components, governing thermal-hydraulic and neutronic behavior and safety systems are analyzed.
- Performance of integral tests and code evaluation of DBA for SBWR-1200: Technical tasks under this milestone are:
 - Obtaining the initial conditions to conduct integral tests on the PUMA facility using REALP5/MOD3 code for the main steam line break (large break), and bottom drain line break (small break) tests
 - Performing integral tests as per well established procedures and quality assurance program
 - Demonstration of the performance of safety systems
 - Examination of the safety analysis code capability, code improvement and validation
- Safety analysis for anticipated transient with scram.
- Core neutronic design: Here the technical tasks are to implement depletion capability to PARCS, perform a preliminary lattice design for the SBWR fuel assemblies, to improve the neutronics/thermal-hydraulics model of all of the SBWR models, and to perform preliminary design analyses of the SBWR-200 and SBWR-1200.
- Analysis of reactor dynamics and instability for SBWR-1200: Under this milestone the technical task is the simulation of normal and abnormal start-up, and instability for the SBWR-1200 using the RAMONA-4B code.

3.3 Third Year Milestones and Technical Tasks

Here the milestones for the third year of the project are presented along with their respective technical tasks.

- Integral effects tests ; multiple failure tests for SBWR 1200 and SBWR-200: Technical tasks under this milestone are:
 - Obtaining the initial conditions to conduct integral tests on the PUMA facility using RELAP5/MOD3 code for the main steam line break (large break), and bottom drain line break (small break) tests
 - Performing integral tests as per well established procedures and quality assurance program
 - Demonstration of the performance of safety systems
 - Examination of the safety analysis code capability, code improvement and validation
- Separate effects tests: Suppression pool dynamics and condensation, and drywell phenomena. In this task the suppression pool condensation capability during the blowdown process are examined. The drywell phenomena include the pressurization during blowdown, depressurization with PCCS condensation and stratification.
- Data Analysis of the integral and separate effects tests, code modeling and scale-up study: Here the task are to analyze the experimental data. Obtain the RELAP5/MOD3 predictions for the integral tests by simulating the prototype reactor design. Compare the PUMA integral tests by scale-up procedure with the prototype predictions.
- Detailed core physics calculations for SBWR-1200 reactor: Here the technical tasks are to implement depletion capability to PARCS, perform a lattice design for the SBWR fuel assemblies, to improve the neutronics/thermal-hydraulics model of all of the SBWR models, and to perform design analyses of the SBWR-1200.
- Analysis of reactor dynamics and instability for SBWR-1200: Under this milestone the technical task is the simulation of normal and abnormal start-up, and instability for the SBWR-1200 using the RAMONA-4B code.
- Final scientific design of SBWR-1200 and SBWR-200: In this milestone the task is to obtain the final design of the SBWR-1200 and SBWR-200 reactors based on thermal hydraulics and neutronic analysis.

4. SCALING ANALYSIS

The objective of this task was to develop a comprehensive scaling methodology that is used in the design of the scaled facility, analysis of the data from the scaled facility, scale-up of the data from scaled model to the prototype and prototype reactor design. In this chapter a detailed scaling analysis is presented. The scaling analysis identifies key thermalhydraulics parameters that govern flow phenomena in SBWR. The analysis deals first with the integral system behavior under both static and dynamic conditions, and then detailed local thermalhydraulics phenomena.

The scaling methodology for the design of the SBWR-200 and SBWR-1200 is based on the three-level scaling approach developed by Ishii et al [4.1]. The scaling consists of the integral system scaling, whose components comprise the first two levels, and the phenomenological scaling which constitutes the third level of scaling. More specifically, the scaling is considered as follows: (1) the integral response function scaling, (2) control volume and boundary flow scaling, and (3) local phenomena scaling. The first two levels are termed the *top-down* approach while the third level is the *bottom-up* approach. This scheme provides a scaling methodology that is practical and yields technically justifiable results. It ensures that both the steady state and dynamic conditions are simulated within each component, as well as the inter-component mass and energy flows, and the mass and energy inventories within each component.

The integral system scaling is derived from the transient response functions for the significant variables in single and two-phase flow. This scaling ensures that the steady state and dynamic conditions are simulated within each component. The integral response function scaling results in the simulation of all the major thermal-hydraulic parameters. To simulate the relations between the various components, the scaling criteria is based on the conservation principles of mass and energy. The control volume balance equations are used to obtain the key scaling criteria for the inter-component relations. To apply this scaling methodology, the integral scaling methods are applied to the system circulation paths. Next, the integral balance and component boundary flow scaling considerations are applied in order to preserve integral mass and energy inventory. The local phenomena such as choking, condensation, flow regimes, etc, are then scaled to preserve similarity and the scaling criteria are obtained..

4.1 Integral System Scaling (1st level)

The various components in the SBWR operate under single phase as well as two-phase flow conditions. Therefore, the overall system scaling should satisfy both the single phase and two-phase flow scaling criteria consistently. In view of this, both the integral system scaling criteria for the single phase and two phase flow are imposed on the system simultaneously. It turned out that the two-phase flow scaling criteria were more restrictive than those for the single-phase flow scaling criteria. Furthermore, it can be shown that the two-phase flow scaling criteria can satisfy the requirements of the single-phase flow scaling criteria. In what follows, the scaling criteria for the single-phase flow and two-phase flow are summarized as a ready reference.

Each component is considered to have a thermal energy source, energy sink and connecting flow path. For a natural circulation loop under single phase flow conditions, the similarity criteria are obtained from the integral effects of the local conservation equations of mass, momentum and energy along the entire flow path.

The fluid continuity, integral momentum, and energy equations in one-dimensional, area-averaged forms are used along with the appropriate boundary conditions and the solid structure energy equation. From these equations, important dimensionless groups characterizing geometric, kinematic, dynamic and energetic similarity parameters are derived. They are as follows:

$$\text{Richardson Number, } R \equiv \frac{g\beta \Delta T_o l_o}{u_o^2} \quad (4.1)$$

$$\text{Friction Number, } F_i \equiv \left(\frac{fl}{d} + K \right)_i \quad (4.2)$$

$$\text{Modified Stanton Number, } St_i \equiv \left(\frac{4hl_o}{\rho_f c_{pf} u_o d} \right)_i \quad (4.3)$$

$$\text{Time Ratio Number, } T_i^* \equiv \left(\frac{l_o / u_o}{\delta^2 / \alpha_s} \right)_i \quad (4.4)$$

$$\text{Heat Source Number, } Q_{si} \equiv \left(\frac{q_s''' l_o}{\rho_s c_{ps} u_o \Delta T_o} \right)_i \quad (4.5)$$

$$\text{Biot Number } B_{ii} = (h \delta / k_s)_i, \quad (4.6)$$

where subscripts i, f and s identify the ith component of the loop, fluid and solid, respectively. Here u_o , ΔT_o and l_o are reference velocity, temperature difference and equivalent length, respectively (for PUMA, l_o is the heated length and ΔT_o is the temperature rise across the core). The symbols appearing in the above set of equations conform to standard nomenclature.

In addition to the physical similarity groups defined above, several geometric similarity groups are obtained as well. These are:

$$\text{Axial Length Scale: } L_i \equiv l_i / l_o$$

$$\text{Flow Area Scale: } A_i \equiv a_i / a_o, \quad (4.7)$$

where a_o is the cross-sectional flow area at the reference component (i.e. chimney).

It is noted here that the hydraulic diameter of the ith section, d_i , and the conduction depth, δ_i , are defined by

$$d_i \equiv 4 a_i / \xi_i \quad (4.8)$$

and

$$\delta_i \equiv a_{si} / \xi_i, \quad (4.9)$$

where a_i , a_{si} and ξ_i are the flow cross sectional area, solid structure cross sectional area and wetted perimeter of the ith section. Hence, d_i and ξ_i are related by

$$d_i = 4(a_i / a_{si}) \delta_i. \quad (4.10)$$

The reference velocity, u_o , and temperature difference, ΔT_o are obtained from the steady-state solution. If the heated section is taken as the representative section, these characteristic parameters are expressed as follows:

$$u_o = \left[\frac{4 \beta g \left(\frac{q_o''' l_o}{\rho_f c_{pf}} \right) \left(\frac{a_{so}}{a_o} \right) l_o}{\sum_i (F_i / A_i^2)} \right]^{1/3} \quad (4.11)$$

and

$$\Delta T_o = \left(\frac{q_o''' l_o}{\rho_f c_{pf} u_o} \right) \left(\frac{a_{so}}{a_o} \right), \quad (4.12)$$

where the subscript o here denotes the heated section and a_{so} is the reference heated surface area. Therefore, u_o and ΔT_o are the natural circulation representative velocity and temperature

rise over the heated section that can be obtained if the system is operated under steady state conditions.

Eqs. (4.1) through (4.5) represent relationships between the dimensionless parameters and the generalized variables characterizing the system under consideration. The similarity criteria between different systems can be obtained through detailed consideration of the similarity groups listed above, together with the necessary closure conditions. If similarity is to be achieved between processes observed in the prototype and in a model, it is necessary to satisfy the following requirements:

$$A_{iR} = (a_i / a_o)_R = 1 \quad (4.13)$$

$$L_{iR} = (l_i / l_o)_R = 1 \quad (4.14)$$

$$\left(\sum_i F_i / A_i^2 \right)_R = \left[\sum_i \left(f_i \frac{l_i}{d_i} + K_i \right) / (a_i / a_o)^2 \right]_R = 1 \quad (4.15)$$

$$R_R = (\beta \Delta T_o l_o / u_o^2)_R = 1 \quad (4.16)$$

$$St_{iR} = (h l_o / \rho_f c_{pf} u_o d_i)_R = 1 \quad (4.17)$$

$$T_{iR}^* = \left[(l_o / u_o) / (\delta^2 / \alpha_s)_i \right]_R = 1 \quad (4.18)$$

$$B_{iR} = (h \delta / k_s)_{iR} = 1 \quad (4.19)$$

$$Q_{siR} = (q_s''' l_o / \rho_s c_{ps} u_o \Delta T_o)_{iR} , \quad (4.20)$$

where the subscript i designates a particular component and R denotes the ratio of the value of a model to that of the prototype, i.e.,

$$\psi_R \equiv \frac{\psi_m}{\psi_p} = \frac{\psi \text{ for model}}{\psi \text{ for prototype}} . \quad (4.21)$$

The frictional similarity requirement, Eq. (4.15), can be satisfied independently of the remaining scaling requirements. Hence, from the remaining scaling requirements, it can be shown that the following conditions should be satisfied for a complete simulation:

$$(u_o)_R = \left(\frac{\beta q_o''' l_o^2}{\rho_s c_{ps}} \right)_R^{1/3} \quad (4.22)$$

$$(\Delta T_o)_R = \left(\frac{q_o''' l_o}{\rho_s c_{ps} u_o} \right)_R \quad (4.23)$$

$$(\delta_i)_R = (\delta)_R = \left(\frac{\alpha_s l_o}{u_o} \right)_R^{1/2} \quad (4.24)$$

$$(d_i)_R = (d)_R = \left(\frac{\rho_s c_{ps}}{\rho_f c_{pf}} \right)_R \left(\frac{\alpha_s l_o}{u_o} \right)_R^{1/2} \quad (4.25)$$

$$(h_i)_R = (h)_R = (k_s)_R \left(\frac{u_o}{l_o \alpha_s} \right)_R^{1/2}, \quad (4.26)$$

where the parameters without the component subscript, i , denote universal values that must be satisfied in all components. In addition to the above, the geometric similarity requirements dictate that

$$\left(\frac{l_i}{l_o} \right)_R = 1 \quad \text{and} \quad \left(\frac{a_i}{a_o} \right)_R = 1 \quad (4.27)$$

must also be met.

With these conditions, the effects of each term in the conservation equations are preserved in the model and prototype without any distortions. If some of these requirements are not satisfied, then the effects of some of the processes observed in the model and prototype will be distorted.

At this point, a few comments are appropriate regarding the practical implications of the similarity requirements:

1. The friction similarity may be difficult to satisfy, except in components having subchannel geometry. Often, friction similarity imposes the most significant limit on the size of a scaled-down model.

2. The conduction depth ratio and hydraulic diameter ratio should satisfy certain criteria.

However, satisfying those criteria over the entire loop may be difficult. It is considered that they are important mainly at the major heat transfer components where these conditions can be easily satisfied. However, the distortions in these criteria over a loop may lead to an overall scale-distortion in terms of structural heat losses. This should be carefully evaluated and compensated.

3. In contrast to the design parameters, the heat transfer coefficient cannot be independent of the flow field. Therefore, there may be some difficulties in meeting the constraint imposed by Eq. (4.26). Satisfying this condition depends on the flow regime. However, relaxation of this similarity requirement influences only the boundary layer temperature drop simulation. When the heat transfer mechanism is not completely simulated, the system will adjust to a different temperature drop in the boundary layer. The overall flow and energy distribution will not be strongly affected during the slow transients typical of a natural circulation system.

4. It is important to note that the above set of requirements does not put constraints on the power density ratio, $\dot{q}_{oR}'''$. However, they do put a restriction on the time scale as follows:

$$\tau_R \equiv \left(\frac{l_o}{u_o} \right)_R = \frac{l_{oR}}{\left[(\beta \dot{q}_o''' l_o^2)_R / (\rho_s c_{ps})_R \right]^{1/3}} . \quad (4.28)$$

The integral system scaling criteria for two phase flow systems have been obtained from the application of the small perturbation technique to the one-dimensional drift flux model by Ishii and Kataoka [4.2]. The four-equation drift flux model consisting of the mixture mass, momentum and energy equations and vapor continuity equation is analytically integrated along the flow path. From this, the integral response functions between various variables such as the velocity, density, void fraction, enthalpy and pressure drop is obtained. The non-dimensionalization of these response functions yields the key integral scaling parameters. From these, the scaling criteria for dynamical simulation can be obtained. The important dimensionless groups that characterize the kinematic, dynamic and energy similarities are given below:

$$\text{Phase Change No.} \quad N_{pch} \equiv \left(\frac{4 \dot{q}_o''' \delta l_o}{du_o \rho_f i_{fg}} \right) \left(\frac{\Delta \rho}{\rho_g} \right) = N_{Zu} . \quad (4.29)$$

This phase change number has been recently renamed to the Zuber number, N_{Zu} , in recognition of Zuber's significant contribution to the field.

$$\text{Subcooling No.} \quad N_{sub} \equiv \left(\frac{i_{sub}}{i_{fg}} \right) \left(\frac{\Delta \rho}{\rho_g} \right) \quad (4.30)$$

$$\text{Froude No.} \quad N_{Fr} \equiv \left(\frac{u_o^2}{g l_o \alpha_o} \right) \left(\frac{\rho_f}{\Delta \rho} \right) \quad (4.31)$$

$$\text{Drift-flux No.} \quad N_{di} \equiv \left(\frac{V_{gj}}{u_o} \right)_i \quad (\text{or Void-Quality Relation}) \quad (4.32)$$

$$\text{Time Ratio No.} \quad T_i^* \equiv \left(\frac{l_o / u_o}{\delta^2 / \alpha_s} \right)_i \quad (4.33)$$

$$\text{Thermal Inertia Ratio} \quad N_{thi} \equiv \left(\frac{\rho_s c_{ps} \delta}{\rho_f c_{pf} d} \right)_i \quad (4.34)$$

$$\text{Friction No.} \quad N_{fi} \equiv \left(\frac{f l}{d} \right)_i \left[\frac{1 + x (\Delta \rho / \rho_g)}{(1 + x \Delta \mu / \mu_g)^{0.25}} \right] \left(\frac{a_o}{a_i} \right)^2 \quad (4.35)$$

$$\text{Orifice No.} \quad N_{oi} \equiv K_i \left[1 + x^{3/2} (\Delta \rho / \rho_g) \right] \left(\frac{a_o}{a_i} \right)^2, \quad (4.36)$$

where α_o , the reference void, in Eq. (4.31) is given by

$$\alpha_o = \frac{\rho_f}{\Delta \rho} = \frac{1}{1 + (N_d + 1) / (N_{Zu} - N_{sub})}.$$

Also, V_{gj} , i_{fg} , i_{sub} , and x are the drift velocity of the vapor phase, heat of evaporation, subcooling and quality, respectively. In addition to the above-defined physical similarity groups, several geometric similarity groups such as (l_i / l_o) and (a_i / a_o) are obtained.

The Froude, friction and orifice numbers, together with the time ratio and thermal inertia groups, have their standard significance. Subcooling, Zuber and drift-flux numbers are associated with the two-phase flow systems.

Eqs. (4.29) through (4.36) represent relationships between the dimensionless groups and the generalized variables of a two-phase flow system. The dimensionless groups must be equal in the prototype and model if the similarity requirements are to be satisfied. Hence, the following conditions result:

$$\begin{aligned} (N_{Zu})_R &= 1, (N_{sub})_R = 1, (N_{Fr})_R = 1, (N_{di})_R = 1 \\ (T_i^*)_R &= 1, (N_{thi})_R = 1, (N_{fi})_R = 1, \text{ and } (N_o)_R = 1. \end{aligned} \quad (4.37)$$

It can be shown from the steady-state energy balance over the heated section that N_{Zu} and N_{sub} are related by

$$N_{Zu} - N_{sub} = x_e \left(\frac{\Delta \rho}{\rho_g} \right), \quad (4.38)$$

where x_e is the quality at the exit of the heated section. Therefore, the similarity of the Zuber and subcooling numbers yields

$$(x_e)_R \left(\frac{\Delta \rho}{\rho_g} \right)_R = 1. \quad (4.39)$$

This indicates that the vapor quality should be scaled by the density ratio. When combined with Eqs. (4.35) and (4.36), Eq. (4.39) shows that the friction similarity in terms of N_{fi} and N_{oi} can be approximated by dropping the terms related to the two-phase friction multiplier. Furthermore, by definition it can be shown that $N_d = (\Delta \rho / (\rho_g x)) [\rho_f / (\Delta \rho \alpha) - 1] - 1$. Therefore, similarity of the drift-flux number requires void fraction similarity

$$(\alpha_e)_R \left(\frac{\Delta \rho}{\rho_f} \right)_R = 1 \quad \text{or} \quad (\alpha_e)_R \approx 1. \quad (4.40)$$

Excluding the friction, orifice and drift-flux number similarities from the set of similarity requirements, Eq. (4.37), and solving the remaining equations, one obtains the following similarity requirements:

$$(u_o)_R = (l_o)_R^{1/2} \quad (4.41)$$

$$(i_{sub})_R = \left(\frac{i_{fg} \rho_g}{\Delta \rho} \right)_R \quad (4.42)$$

$$(q_o''')_R = \left(\frac{\rho_f \rho_g i_{fg}}{\Delta \rho} \right)_R \left(\frac{d}{\delta} \right)_R (l_o)_R^{-1/2} \quad (4.43)$$

$$\delta_R = (l_o)_R^{1/4} (\alpha_s)_R^{1/2} \quad (4.44)$$

$$d_R = \left(\frac{\rho_s c_{ps}}{\rho_f c_{pf}} \right)_R (l_o)_R^{1/4} (\alpha_s)_R^{1/2}. \quad (4.45)$$

The velocity scale shows that, in contrast to the case of single-phase flow scaling, the time scale for a two-phase flow is not an independent parameter. From Eq. (4.41), the time scale in two-phase flow is uniquely established, thus,

$$\tau_R = \left(\frac{l_o}{u_o} \right)_R = (l_o)_R^{1/2} . \quad (4.46)$$

This implies that if the axial length is reduced in the model, then the time scale is shifted in the two-phase flow natural circulation loops. In such a case, the time events are accelerated (or shortened) in the scaled-down model by a factor of $(l_{oR})^{1/2}$ over the prototype. It is important to note that when the two phase flow velocity scale, Eq. (4.41), is used in the single phase flow geometric scale requirements, the geometric similarity requirements in both cases become the same. Hence, the same geometric scale can be used for single phase and two phase flows. However, using the time scale indicated by the two phase flow scaling, namely $\tau_R = \sqrt{l_{oR}}$, the single phase time events are also scaled by the same criterion. This leads to the very important conclusion that for systems involving both single and two phase flow in a reduced length model, real-time scaling is not appropriate.

The similarity criteria between different systems can be obtained through detailed consideration of the similarity groups listed above, together with the necessary closure conditions. If similarity is to be achieved between the processes observed in the prototype (the reference reactor SBWR-600) and the model (new design SBWR-200 or SBWR-1200), the property should be defined as follows:

$$\psi_R \equiv \frac{\psi_m}{\psi_p} = \frac{\psi \text{ for model}}{\psi \text{ for prototype}} . \quad (4.47)$$

Similarity is then achieved if the following requirements are satisfied: $\psi_R = 1$ for all the similarity parameters.

4.2 Mass and Energy Inventory and Boundary Flow Scaling (2nd Level)

A nuclear reactor system such as the SBWR consists of several inter-connected components. Therefore, it is essential to simulate the thermal-hydraulic interactions between these components. The physical processes involved in the system are governed by the conservation principles of mass, momentum, and energy. Among them, both the mass and energy balances are key to the proper scaling of the inter-component relations. The conservation of momentum is important for the forces acting on the structure, however, it is not essential to the scaling of the inter-component thermal-hydraulics. For a system consisting of several components, the scaled

mass and energy inventory histories must be preserved for the integral similarity of the thermodynamic state of each component. The conservation of momentum becomes important in determining the boundary mass flow.

The scaling criteria can be obtained from the control volume balance equations for mass and energy. In particular, important scaling criteria are obtained for the boundary flow of mass and energy at the interface between two connected components. As is the case of many types of transients, both choked and non-choked flow can occur at the same junction. If the non-choked flow is governed by the frictional resistance, it can be scaled by the integral scaling criteria based on the response function. However, during the blowdown phase, the choked flow is dominated by non-frictional momentum effects. At such discharge points, the fluid velocity depends upon the local pressure ratio across the device, which is preserved in a full-pressure scaled system. In non-frictional momentum-dominated flows, the fluid velocity is the same in the model as in the prototype. Therefore, the flow area at such discharge points must be scaled to preserve mass and energy inventory rather than loop kinematics. Under this scaling the appropriate scaling relations are developed. An overall criterion for similar behavior between the prototype and the model is that the depressurization histories be the same when compared in the respective (scaled) time frames, i.e.,

$$p_m(t_m) = p_p(t_p). \quad (4.48)$$

This integral condition will be satisfied if the differential pressure change is the same at corresponding times, i.e.,

$$\frac{dp_m}{dt_m} = \frac{1}{\tau_R} \frac{dp_p}{dt_p}. \quad (4.49)$$

The scaling criteria for similarity of the friction-dominated natural circulation flows, yields the result that the time scale of the model or laboratory time, is related to the prototype time by:

$$t_m = (l_o)_R^{\frac{1}{2}} t_p = \tau_R t_p. \quad (4.50)$$

and the depressurization rates of the model and the prototype are related by:

$$\frac{dp_m}{dt_m} = \left(\frac{1}{I_o} \right)_R^{\frac{1}{2}} \frac{dp_p}{dt_p}. \quad (4.51)$$

This condition will be satisfied if the corresponding component vessel inventories are similar, i.e.,

$$\left(\frac{M_m}{V_m} \right)_{t_m} = \left(\frac{M_p}{V_p} \right)_{t_p}. \quad (4.52)$$

where M_p and M_m are the prototype and model vessel inventory masses, and V_p and V_m are the respective prototype and model vessel volumes. This relation must hold for each component as well as for the overall system if complete similarity is to be ensured.

For integral experiments, accurate simulation of the mass and energy inventory is essential. This requires a separate scaling criteria for the system boundary flows such as the break flow and various ECCS injection flows. The similarity condition for the flow area and velocity are

$$\left(\frac{a_{in}}{a_o} \frac{u_{in}}{u_o} \right)_R = 1, \text{ and } \left(\frac{a_{out}}{a_o} \frac{u_{out}}{u_o} \right)_R = 1. \quad (4.53)$$

The similarity requirement on the enthalpy $(i_{in})_R = 1$ and $(i_{out})_R = 1$, implies that the inflow or outflow should have a prototypic enthalpy. The initial energy inventory should be scaled by the volume ratio. Under a prototypic pressure simulation, the system geometry can be determined from the integral system scaling and the boundary flow scaling. Reduce height-scaling shows certain advantage.

4.3 Local Phenomena Scaling (3rd Level)

In this level scaling the following phenomena are considered.

(1) Reactor vessel flow dynamics and instability scaling, (2) Break and ADS flow scaling, (3) Scaling for flows driven by elevation change, (4) Relative velocity and flow regime, (5) Critical heat flux scaling (CHF), (6) Flashing in the chimney, (7) Condensation in suppression pool, (8) Vent phenomena in suppression pool, (9) Mixing in stratified fluid volumes, (10) Natural circulation, (11) Heat source and sink, (12) PCCS venting into suppression pool, (13)

Condensation in PCCS condensers, (14) Stratification in the drywell and (15) Stratification in the suppression pool.

Figure 4.1, summarizes the scaling methodology developed for the design of the PUMA facility. The scaling laws developed here are applied in the integral tests data analysis and design of the SBWR-200 and SBWR-1200 systems.

4.4 References

- 4.1 Ishii, M., Revankar S. T., Dowlati, R., Bertodano, M. L., Babelli, I., Wang, W., Pokharna, H., Ransom, V. H., Viskanta, R., Wilmarth, T., and Han, J. T., “Scientific Design of Purdue University Multi-Dimensional Integral Test Assembly (PUMA) for GE SBWR,” Purdue University Report PU-NE-94/1, U.S. Nuclear Regulatory Commission Report NUREG/CR-6309, 1996.
- 4.2 Kataoka I., Ishii M., Drift Flux Model for Large Pipe and New Correlation for Pool Void Fraction, *Internat. J. Heat Mass Transfer*, Vol. 30. 1927-1939, 1987.

Figure 4.1 PUMA scaling methodology chart

5. DESIGN OF SBWR-200 AND SBWR-1200

The objective of this task was to develop design improvements to the SBWR-200 and SBWR-1200. The design involves identification of the principle design criteria dictated by the safe operation of the reactor, identification of coolant requirements, design of the engineered safety systems, and emergency cooling systems based on passive systems. The SBWR-200 was designed to be modular. The large SBWR-1200 design has all essential safety systems. In addition to these, a new design of vacuum breaker check valve, which is based on hydraulic head, was developed. This valve is passive unlike the previous design, which is a mechanical valve. The passive vacuum breaker check valve performance was first evaluated using RELAP5. Then the passive vacuum breaker check valve was constructed and implemented in the PUMA facility. Its performance was tested during a LOCA transient.

5.1 Principal Design Criteria:

The design of SBWR-200 and SBWR-1200 was based on the extrapolation of the General Electric SBWR-600 design [5.1]. The principal design criteria governing the SBWR were presented in two ways. First, the criteria were classified as applicable to either a power generation function or a safety-related function. Second, they were grouped according to system. Here some of the relevant criteria for the scientific design of the two SBWR plants are listed below:

Power Generation Design Criteria:

- 1) 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.
- 2) Backup heat removal systems are provided to remove decay heat generated in the core under conditions when the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel damage.
- 3) Passive safety systems are provided to mitigate core uncover and overheating during abnormal conditions such as loss of coolant accident and other transients.

Nuclear System Criteria:

- 1) The capacity of the heat removal systems provided to remove heat generated in the reactor core for 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.
- 2) The reactor is capable of being shutdown automatically in sufficient time to prevent decay heat sinks to become effective following loss of operation of normal heat removal systems.

Nuclear Fuel design Consideration:

The key characteristics of the established BWR fuel technology are used in the SBWR design. These are: (1) uranium oxide based fuel pellets; (2) zirconium-based fuel cladding; (3) material selected on the basis of BWR operating conditions; (4) multi-rod fuel bundles; and (5) fuel bundle inlet orificing to control bundle flow rates, core flow distribution, and reactor coolant hydraulic characteristics. It should be noted that if natural circulation cooling is used core should be designed with low power density.

One of the important criteria in the design of the fuel bundle is to maintain continuous coolability under all conditions. This ensures the thermal-mechanical integrity of the fuel bundle. The nuclear characteristics considered in the design of the fuel bundles are (1) a negative Doppler reactivity is maintained; (2) A negative core moderator void reactivity coefficient resulting from boiling in the active flow channel is maintained for design basis operating conditions; (3) a negative moderator temperature reactivity coefficient is maintained above hot standby; (4) for a super critical reactivity accident, the net prompt reactivity feedback due to prompt heating of the moderator and fuel is negative; and (5) adequate plant cold shutdown margin is maintained.

5.2 Safety Requirements for Design Basis Accident

The safety systems are defined based on the requirement of safety against design basis accidents (DBA). These are listed here.

5.2.1 General Safety Design Criteria:

- 1) The reactor core is designed such that its nuclear characteristics do not contribute to a divergent power transient.

- 2) Cooling of the reactor core is assured following anticipated reactor accidents.
- 3) Safety-related systems and engineered safety features function to assure that no damage to the reactor coolant pressure boundary results from internal pressure caused by abnormal operational transients and accidents.
- 4) Precise automatic actions are taken in response to abnormal operational transients and accidents.
- 5) Safety-related systems are designed to permit demonstration of their functional performance requirements.
- 6) Containment employs the pressure suppression concept.
- 7) The areas above the containment top slab and drywell head are flooded in a pool of water during operation.
- 8) 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.
- 9) Emergency core cooling is provided to limit fuel clad temperature to less than limits of 10CFR50.46 in the event of a design basis loss-of-coolant accident (LOCA).
- 10) The emergency core cooling provides for continuous core cooling over the complete range of postulated break sizes in the reactor pressure boundary piping.
- 11) Emergency core cooling is initiated automatically when required, regardless of the availability of off-site power supplies and the availability of the normal generating system of the station.

5.2.2 Emergency Core Cooling Systems:

The GE-SBWR emergency core cooling systems (ECCS) are the Gravity-Driven Cooling System (GDCS) and the Automatic Depressurization System (ADS) [5.1]. The ECCS is designed to provide protection against postulated loss-of-coolant accidents (LOCA) caused by the rupture of the primary system piping. It is designed to meet the following requirement: (1) Protection is provided for any primary system line break up to and including the double ended break of the largest line. (2) No operator action is required until 72 hours after an accident. (3) A sufficient water source and necessary piping is provided so that the containment and reactor core can be

flooded for possible core heat removal following LOCA. For the long term cooling, the decay heat is removed by the passive containment cooling system (PCCS).

5.3 SBWR Safety Systems

The engineered safety systems and safety-grade systems in the SBWR include the following: (1) Automatic Depressurization System (ADS), (2) Gravity-Driven Cooling System (GDCCS), (3) Passive Containment Cooling System (PCCS), and (4) Isolation Condenser Systems (ICS). The GDCCS and PCCS are new designs unique to the SBWR and do not exist in currently operating BWRs. The ICS is functionally similar to those in some current BWRs. Both the GDCCS and PCCS are designed for low-pressure operation (less than 1033 kPa or 150 psia), but the ICS is capable of high-pressure operation as well (up to 7580 kPa or 1100 psia). The ADS will be actuated at a prescribed vessel condition and depressurizes the reactor vessel so that the gravity driven cooling systems can be activated which leads to water injection. The SP water is available to flood the reactor core. Thus, these safety systems satisfy all the requirements for design basis accident (DBA).

5.4 Design Parameters for SBWR-200 and SBWR-1200

Based on the scaling criteria the design of SBWR-200 and SBWR-1200 has been performed. In this design it was assumed that the new design of the reactor operate at the same pressure as the reference reactor SBWR-600.

At the prototypic (reference reactor) pressure simulation, the following relations are obtained from the integral system scaling and the boundary flow scaling results given in preceding section

$$p_R = 1$$

$$A_{iR} = \left(\frac{a_i}{a_o} \right)_R = 1$$

$$L_{iR} = \left(\frac{l_i}{l_o} \right)_R = 1$$

$$u_R = v_R / l_R^{1/2}$$

$$q_R = \frac{v_R}{l_R^{1/2}}$$

At the major heat transfer sections, such as at the core and heat exchanges, it may be possible to make $d_R = 1$, which implies that the heater or heat exchanger sections should have a prototypic hydraulic diameter.

The major portion of the design information on the simplified boiling water reactor is collected from the Standard Safety Analysis Report from General Electric on 600MWe SBWR [5.1]. The design of the GE-SBWR was based on the proven technology of GE obtained during the development, design, construction and operation of BWR. In Table 5.1, the reactor system design characteristics of ABWR and SBWR are listed.

In the design of the SBWR-200 and SBWR-1200, the operating reactor pressure was assumed as the reference reactor operating pressure. The heights of the reactor including reactor vessel, containment, suppression pool, emergency core cooling systems and the condenser systems were assumed to be same as the reference reactor design. In Table 5.2, the thermohydraulic characteristics, fuel characteristics and other reactor components are listed with the design parameters. The ratio of the component parameter with reference parameter is also given in this table to indicate the scaling criteria used in deriving the design parameters.

5.5 Design Characteristics of SBWR:

In this section the principal design features of the SBWR-200 is discussed. The design considerations for the SBWR-1200 are similar. As described in previous chapters that the SBWR is has simplified primary coolant system compared with BWR. All the current well established BWR technology is still applicable to this reactor. The recirculation pumps are eliminated and the core cooling is provided by natural circulation. The core has low power density. The reactor design is based on passive emergency cooling systems. Thus no emergency AC power required for DBA. However there are technical challenges such as maintenance of steady flow during natural circulation cooling, and avoidance of core flow and power instabilities during normal operation and during transients such as start-up and shutdown.

The primary and associated cooling systems for the reactor under consideration are briefly discussed here. The containment boundary of the reactor has reactor pressure vessel (RPV), DW, SP, GDCS, isolation condenser (ICS) piping, PCCS piping and ADS. The condensers and pools for the ICS and pools for PCCS are located outside and above the

containment boundary. The reactor has a unique depressurization scheme where the ADS system is activated when the RPV water level goes below certain specified value. The ADS, which consists of six depressurization valves (DPVs) and six-steam relief valves (SRV), open in sequence when activated as given in the following Table 5.3. SP provides pressure suppression during initial blow down steam load by direct contact condensation. At the time of ADS activation the injection valve of GDCS are activated to open. When the RPV pressure levels with DW/GDCS cover pressure, the GDCS injects cold water directly into RPV. Any further decrease of water level in RPV, when still higher than the top of active fuel triggers the water injection from SP through GDCS equalization lines to RPV. Thus core uncover is totally avoided. For long term cooling the PCCS condenses steam from the containment environment and returns the water to RPV. All the core cooling mechanisms are based on the gravity force and hence ensure safety and reliability.

5.6 Steady State Coolant Requirement for SBWR-200:

For the design of the 200 MWe SBWR referred as SBWR-200, the design parameters as given in Table 5.2. Using this information the steady state design calculations were performed for the proposed SBWR-200 reactor. The reference design thermal output power of the proposed SBWR-200 is assumed to be at one third of the thermal power of the SBWR-600. Thus the output thermal power of SBWR-200 is taken at 660 MWt. The core inlet enthalpy is calculated as $h_{in} = h_f - c_p(T_f - T_{in})$. The core exit enthalpy is calculated as $h_{out} = h_f + x_{ave} h_{fg}$. The coolant flow rate is thus given as $\dot{m}_{core} = (h_{out} - h_{in}) / (h_{fg} - h_f)$. The feed water flow rate is calculated as $\dot{m}_{FW} = x_{ave} \dot{m}_{core}$. The steam flow rate is take equal to the feed water flow. The coolant flow rate per rod in the SBWR-600 is $\dot{m}_{rod} = \dot{m}_{core} / N_{rod}$, where N_{rod} is the total number of rods in the core. For the SBWR-200, the fuel rod geometry in the fuel bundle is taken to be the same as in the SBWR-600 and the average linear heat rate and average heat flux per rod are as given in Table 5.2. Thus the average mass flow rate per rod in the SBWR-200 is the same as in the SBWR-600. The total number of fuel rods for the SBWR-200 is then calculated as the ratio $\dot{m}_{core} / \dot{m}_{rod}$. Using the number of rods and the total power of 660 MWt, the power per rod in SBWR-200 is calculated as $Q_{rod} = 660 \text{ MW} / (\text{number of rods})$. For the same average linear heat generation rate q'_{ave} as in the SBWR-600, the length of the SBWR-200 fuel rod is calculated as $L_{rod} = Q_{rod} / q'_{ave}$.

Assuming the number of fuel rods in each assembly as 60 (as in the case of SBWR-600), the number of fuel assemblies is calculated as (number of rods)/60. As a first approximation, the diameter of the core is proportional to the number of assemblies. The core diameter for the SBWR-200 is calculated as the ratio of the number of fuel assemblies in SBWR-600 and in SBWR-200 times the core diameter of SBWR-600. From Table 5.4, it is seen that for the SBWR-200 the values of the core power, core coolant flow rate, steam flow rate and the core diameter are about 1/3 of the SBWR-600. The natural circulation drives the SBWR reactor coolant flow. The preliminary steady state reactor coolant requirement clearly shows that the RPV of the SBWR-200 has the same core height as the SBWR-600. Hence, if the average flow quality is the same for both reactor designs, the driving head available at the core for natural circulation flow is the same in both the SBWR-200 and in the SBWR-600. If the chimney height and other flow paths in the SBWR-200 are assumed to be the same as in the SBWR-600 and the flow area at respective sections of the SBWR-200 RPV are 1/9 of the SBWR-600 RPV, then the pressure drop across the core and the other components should be same in both reactors. The reduced flow area will give the reduced flow rate in each components of the SBWR-200.

5.7 Design of SBWR-200 and SBWR-1200

The scaling analysis of the SBWR-200 and SBWR-1200 has been performed. The GE-SBWR was used as a reference in the design of SBWR-200 and SBWR-1200 reactors. The operating reactor pressure of the GE 600 MWe design was assumed as the reference operating pressure. The heights of the reactor components including reactor vessel, containment, suppression pool, emergency core cooling systems and the condenser systems were assumed to be same as the reference design.

The primary and associated cooling systems for the reactor under consideration are briefly discussed here. The containment boundary of the reactor includes the reactor pressure vessel (RPV), DW, SP, GDCCS, isolation condenser (ICS) piping, PCCS piping and ADS. The condensers and pools for the ICS and PCCS are located outside and above the containment boundary. The reactor has a unique depressurization scheme in which the ADS system is activated when the RPV water level goes below certain specified value. The ADS, which consists of six depressurization valves (DPVs) and six-steam relief valves (SRV), open in

sequence when activated. SP provides pressure suppression during initial blow down steam load by direct contact condensation. At the time of ADS activation, the injection valves of the GDCS are opened. When the RPV pressure levels with DW/GDCS pressure, the GDCS injects cold water directly into RPV. Any further decrease of water level in the RPV, triggers the water injection from SP through the GDCS equalization lines. Thus core uncover is totally avoided. For long term cooling the PCCS condenses steam from the containment environment and returns the water to the RPV. All the core-cooling mechanisms are based on gravitational head, and hence ensure safety and reliability.

In Figures 5.1- Figure 5.2, the design of the SBWR-200 reactor pressure vessel and containment system layouts are shown. The pressure vessel wall thickness for SBWR-200 is calculated as 0.18 m to handle the system operation pressure of 7.2 MPa. The internal detail and location of various piping to the reactor vessel are also indicated in Figure 5.1 Due to its modular nature, the containment design is made of steel, which has a bottle shape, as shown in Figure 5.2. The containment wall thickness is 0.02 m. The ICS and PCCS condensers are located on top of the containment structure in partitioned compartments. The steel tank concept allows factory manufacture of the containment and the RPV. The steel tank can be easily transported to the reactor site. The RPV can be installed on site once the containment is housed.

In Figures 5.3 and 5.4, the design of the SBWR-1200 reactor vessel, and the containment system layouts are shown. The reactor vessel wall thickness is 0.4 m. Figure 5.3 also indicates the reactor vessel internals and the position of various piping connections to the reactor vessel. The containment vessel design is similar to the SBWR-600 reactor design, and is constructed of pre-stressed nuclear grade concrete.

5.8 Vacuum Breaker Check Valve

The PCCS is the key component in the long term cooling of the SBWR containment. The PCCS condensers condense steam in the DW. The condensate is then drained to the RPV through the GDCS tank. Non-condensable gases accumulated in the PCCS condensers are vented to the SP gas space. The driving head for the steam is provided by the pressure difference between the DW and the SP. The gas space above the SP serves as the gas reservoir for the nitrogen and other non-condensable gases. During blowdown, most of the non-

condensable gases from the drywell are passed through eight DW-to-suppression chamber (SC) vertical vent pipes to the SP gas space. If the DW steam is condensed, the SP pressure can be higher than the DW pressure. If the SC gas pressure is higher than the upper DW space then the PCCS cannot vent the non-condensable gas. This will drastically reduce the effectiveness of the PCCS in long term containment cooling. To prevent the over pressurization of the SP relative to the DW, there is a vacuum breaker system between the SC and the DW. The vacuum breaker consists of check valves, which open when the SC pressure exceeds the DW pressure at a preset difference. The vacuum breaker check valve, being a mechanical valve, is prone to failure. Failure of this mechanical valve drastically reduces the PCCS cooling capability and hence containment pressure may exceed the design pressure. The vacuum breaker check valves are the only key safety components which are not passive in nature. To eliminate this problem a new design of the vacuum breaker check valve was developed to replace the mechanical valve. This new design is based on a passive hydraulic head. This new design is fail-safe, as it is truly passive in operation. Moreover this new design needs only one additional tank and one set of piping each to the SP and DW. This system is simple in design and hence is easy to maintain and qualify for operation.

5.8.1 Design of Passive Hydraulic Vacuum Breaker Check (HVBC) Valve

The principle of operation of the new passive hydraulic vacuum breaker check (HVBC) valve is based on the free surface of the gas liquid system and the hydrostatic head. There are no moving parts in this design. In Figure 5.5 the schematic of the HVBC valve is compared with the mechanical vacuum breaker check (MVBC) valve. In the MVBC valve, the lifting of the ball occurs when the inlet pressure is higher than the outlet pressure. The mass of the ball determines the differential pressure required to lift the ball. In the GE-SBWR design the pressure difference required to lift the ball is 3.4 kPa. Since the ball physically moves, there is a probability that the ball may not be seated properly after the pressure differences between the outlet and inlet is reduced. If some foreign object is lodged at the ball seat then the check valve will leak. The new design of the HVBC shown in Figure 5.5 consist of a water tank and two ports connecting to the upper drywell and suppression pool gas space. The inlet piping is immersed in the tank water, with the height of the water providing the required head for the flow to commence from inlet to outlet. Change in the water level changes the differential pressure required for the flow to ensue.

However in the mechanical valve the ball has to be replaced for different differential pressure for the valve opening.

To implement this new design in the prototype reactor containment, the following issues were considered. (1) First, if the outlet side pressure is higher (drywell side) than the inlet side, then the water will rise through the submersed piping to the inlet side (suppression pool gas side). This will in effect reduce the tank water level if the water surface area is smaller. Also if the inlet piping is too short, then the water from the tank will drain to the inlet side and thus will fail to provide any hydrostatic head. In order to avoid this, a sufficient height to the inlet piping is provided. The vertical height of the pipe is such that the hydrostatic head associated with the vertical length is larger than the possible maximum pressure difference between the drywell and suppression pool gas chamber. This prevents overflow of the water from the tank to the inlet side. Also, the surface area of the water in the tank is large such that the water rise in the vertical pipe does not change the water level by more than 1%. (2) Since the inlet flow introduces bubbles or gas jet in the tank water, there may be carry over of the entrained droplets to the outlet side. In order to eliminate the droplet carry over, the venting flow area of the submersed pipe should be large. A sparger design is included in the submerged pipe end to reduce the vent gas velocity close to the bubble rise velocity. (3) Since the HVBC valve system is connected to the drywell atmosphere through piping, there is the possibility of continuous evaporation of the water into the containment atmosphere. In order to avoid this a 90° pipe bend is provided and the end of the pipe is at the tank water level. This arrangement minimizes escape of water vapor by mass diffusion from the tank to the containment, by reducing the concentration gradient.

Scaling and design calculations were performed to obtain the geometry of the HVBC valve. The calculations for the SBWR-200, SBWR-1200 and PUMA facility are shown in Table 5.5. This table gives the actual design sizes for the tank, piping and sparger of the HVBC valve. In Figure 5.6 the schematic design of the HVBC valve is shown for the SBWR-200. In Figure 5.7 the prototype design of the SBWR-200 containment is shown with the locations of the tank, and piping of the HVBC valve.

5.8.2 RELAP5 Simulation of HVBC Valve

In order to check the HVBC valve performance, a RELAP5 model of the SBWR-200 reactor system with the HVBC valve was developed. This was done before the design was implemented

on the scaled PUMA facility. A RELAP5 model was developed for the DBA scenario. A large break LOCA, main steam line break accident was simulated for the SBWR-200, using the new HVBC valve and the old MVBC valve. The transient was simulated for a 16 hour period. The predicted results for the HVBC valve and the MVBC valve were compared.

In Figure 5.8 the DW pressure comparison is shown. The agreement between the two predictions is within 5% of the DW pressure. Also it can be seen that long term (at 16 hour) DW pressure is practically the same. The DW pressure decreases as the RPV pressure is decreased due to the injection of emergency water into the RPV, which collapses the void. During this period, the vacuum break check valve opens, as the SP pressure is higher than the DW pressure. The HVBC valve performance is very similar to the MVBC valve. In Figure 5.9, the tank water level is shown during the GDCS injection period. The tank water level remains constant during this period. However the water level in the pipe shows a rise during the blowdown phase when the DW pressure is high. This level then settles back to the normal level after the SP and DW pressure are equalized following the venting.

5.8.3 PUMA HVBC Valve

Based on the scaling and design parameters given in Table 5.5 the PUMA HVBC valve was designed. In Figure 5.10 the design of the PUMA HVBC valve is shown. The tank is provided with viewing windows, drain line and fill line. During the PUMA test the bubble venting process and the water surface in the tank can be observed with these viewing windows. The tank is made of a stainless steel cylinder of diameter 0.61 meter. The sparger sizes and the tank cross section data are given in Table 5.5. The pictures of the PUMA HVBC valve tank and sparger are shown in Figure 5.11. This system has been installed in the PUMA facility. For testing, either this new design of the vacuum breaker check valve or the old mechanical vacuum breaker check valve can be used.

5.8.4 Testing of the PUMA HVBC Valve

The PUMA HVBC valve was tested for a DBA scenario. Two MSLB accident test were carried out on PUMA facility, one with the HVBC valve and the other with the MVBC valve. The tests were carried out for a period of 8 hours. The test results were then compared with each other.

In Figure 5.12, the RPV pressure comparison is shown. The RPV pressure for the test with the HVBC valve agrees very well with the test with the MVBC valve. In Figure 5.13 the drywell pressure comparison is shown. The drywell pressure for the test with the MVBC valve shows a slightly higher pressure compared to the test with the HVBC valve during the first 10,000 seconds. This period corresponds to the emergency water (GDCS) injection, and reheating of the reactor water. However in the long term, the drywell pressures compare very well. This indicates that the overall containment pressure level reached is similar with either type of valve system. The comparison of the collapsed water level in the RPV is shown in Figure 5.14. The agreement between the two cases is within 2%. The RPV water level and the containment are two key parameters related to the safety of the reactor during, and following the accident. The HVBC valve performance in maintaining these two parameters for the MSLB accident scenario is very similar to that with the MVBC valve.

5.9 References

- 5.1 GE Nuclear Energy, "SBWR Standard Safety Analysis Report," Report No. 25A5113 Rev. A, August 1992.

Table 5.1 GE-SBWR and ABWR Design Characteristics

Design Characteristic	Units	GE-SBWR	ABWR
Thermal and Hydraulic			
Reference design thermal power	MWt	2000	3926
Design power (ECCS design basis)	MWt	2084	4005
Steam flow rate, at 215.6 °C final feed water temperature (FFWT)	kg/h	3.89×10^6	7.64×10^6
Core coolant flow rate	kg/h	27.2×10^6	52.2×10^6
Feed water flow rate	kg/h	3.876×10^6	7.17×10^6
Absolute pressure, nominal in steam dome	MPa	7.171	7.171
Absolute pressure, nominal in core	MPa	7.239	7.274
Coolant saturation temperature at core	°C	288.3	288.3
Average power density	kW/liter	41.5	50.6
Average linear heat generation rate	kW/m	16.6	20.3
Average heat flux	kW/m ²	430.58	524.86
Coolant enthalpy at core inlet	kJ/kg	1228.3	1227.2
Core inlet temperature at 215.6 °C FFWT	°C	278.5	278.5
Core average void fraction, active coolant		0.434	0.408
Core average exit quality		14.3	14.5
Total core pressure drop	MPa	0.048	0.168
Core support plate pressure drop	MPa	0.026	0.138
Fuel			
Water/UO ₂ volume ratio (cold)		3	2.95
Doppler coefficient	Cents/°C	-0.39	-0.36
Initial average U ²³⁵ enrichment	%	3.95	2.5
Fuel Assembly			
Number of fuel assemblies		732	872
Fuel rod array size		8x8	8x8
Pitch of square rod array	mm	16.2	16.2
Fuel Rods			
Number of fuel rods per assembly		60	62
Outside diameter	mm	12.27	12.27
Cladding thickness	mm	0.8126	0.864
Cladding material		Zircaloy-2	Zircaloy-2
Core Assembly			
Core diameter	mm	4730	5164
Active fuel length	mm	2743	3078

Table 5.2 Design Parameters for SBWR-200 and SBWR-1200

Nominal in steam dome $P_{RPV} = 7.171$ MPa
Coolant saturation temperature at core $T_f = 288.3$ °C
Average linear heat generation rate $q'_{ave} = 47.2$ kW/m
Average heat flux $q''_{ave} = 430.58$ kW/m ²
Core inlet temperature at 215.6 °C FFWT $T_{in} = 278.5$ °C
Core average void fraction, active coolant $\alpha_{ave} = 0.434$
Core average exit quality $x_{ave} = 14.3$
Fuel rod array size = 8x8
Pitch of square rod array $P_i = 16.2$ mm
Fuel Rod Outside diameter $d_o = 12.27$ mm

Table 5.3 Automatic Depressurization Systems (ADS)

1st 4 SRVs (At activation)	0s
2nd 4 SRVs	10 s
1st 2 DPVs (MSL)	55s
2nd 2 DPVs (RPV)	100s
3rd 2 DPVs (RPV)	145s

Table 5.4 Design and Scaling Characteristics for SBWR-200

Design Characteristic	Units	GE-SBWR	SBWR-200	Ratio
Thermal and Hydraulic				
Reference design thermal power	MWt	2000	660	3.03
Steam flow rate	kg/h	3.89×10^6	1.2758×10^6	3.05
Core coolant flow rate	kg/h	27.2×10^6	8.922×10^6	3.05
Feed water flow rate	kg/h	3.876×10^6	1.2758×10^6	3.04
Absolute pressure, nominal in steam dome	MPa	7.171	7.171	1
Coolant saturation temperature at core	°C	288.3	288.3	1
Average linear heat generation rate	kW/m	16.6	16.6	1
Coolant enthalpy at core inlet	kJ/kg	1228.3	1228.3	1
Core inlet temperature at 215.6 °C FFWT	°C	278.5	278.5	1
Core average exit quality		14.3	14.3	1
Fuel Assembly				
Number of fuel assemblies		732	240	3.05
Fuel rod array size		8x8	8x8	-
Pitch of square rod array	mm	16.2	16.2	1
Fuel Rods				
Number of fuel rods per assembly		60	60	1
Outside diameter	mm	12.27	12.27	1
Core Assembly				
Core diameter	mm	4730	1577	3
Active fuel length	mm	2743	2750	1

Table 5.5 Scaling and design of hydraulic vacuum breaker check valve

Global Scaling

	Unit	PUMA	SBWR-200	Ratio	SBWR-1200	Ratio
Drywell volume	m ³	13.76	1830	133	11008	800
SP gas volume	m ³	9.548	1270	133	7638	800
SP water volume	m ³	8.138	1082	133	6510	800
Vacuum Breaker DP	kPa	3.452	3.452	1	3.452	1
Flow Area				33.3		200
Height				4		4

Sparger Scaling

Sparger design	Unit	PUMA	SBWR-200	SBWR-1200
Limit of discharge velocity	m/s	7	7	7
Pipe flow area	m ²	0.008	0.203	1.216
Pipe diameter	cm	10.16	50.8	124.43
K factor (Minor Loss)		5	5	5
DP between SP and DW	kPa	3.452	3.452	3.452
Maximum DP	kPa	6.904	6.904	6.904
Flow velocity in pipe	m/s	33.89	33.89	33.89
Flow rate in pipe	m ³ /s	0.275	6.87	41.22
No. of holes		490	756	4536
Diameter of holes	cm	1.48	5.08	5.08
Flow area of holes	m ²	0.084	1.532	9.192
Velocity at sparger holes	m/s	3.26	4.483	4.483

Tank and Piping Scaling

	Unit	PUMA	SBWR-200	SBWR-1200
Piping height	m	2.25	9	9
Piping Flow Area	m ²	0.008	0.204	1.216
Piping Volume	m ³	0.018	1.824	10.94
Sparger submerge depth	cm	35.13	35.13	35.13
Tank cross area	m ²	0.465	15.58	93.46
Tank height	m	0.61	2.6	2.6

Figure 5.1 . SBWR-200 reactor pressure vessel and internal components

Figure 5.2 SBWR-200 reactor containment

Figure 5.3 SBWR-1200 reactor pressure vessel and internal components

Figure 5.4 SBWR-1200 reactor containment

Figure 5.5 Operational Principles of Mechanic and Hydraulic Vacuum Breaker Check Valves

Figure 5.6 Schematic of Hydraulic Vacuum Breaker Check Valve Design

Figure 5.7 SBWR-200 Reactor Containment with Hydraulic Vacuum Breaker Check Valve System

Figure 5.8 RELAP5 Simulations of SBWR-200 MSLB with MVBC and HVBC valves, Drywell Pressure

Figure 5.9 RELAP5 Simulations of SBWR-200 MSLB with MVBC and HVBC valves, Water level in tank and pipe

Figure 5.10 Schematic of PUMA Hydraulic Vacuum Breaker Check Valve Design

(a). PUMA Hydraulic Vacuum Breaker Tank

(b). Sparger

Figure 5.11 Pictures of PUMA HVBC Valve System: (a). Tank, (b). Sparger

Figure 5.12 Comparison of PUMA MSLB with MVBC and HVBC valves, RPV pressure

Figure 5.18 Comparison of PUMA MSLB with MVBC and HVBC valves,
Drywell pressure

Figure 5.13 Comparison of PUMA MSLB with MVBC and HVBC valves, drywell pressure

Figure 5.14 Comparison of PUMA MSLB with MVBC and HVBC valves, RPV collapsed water

6. SAFETY STUDY

The objective of this task was to analyze the performance of the safety systems during a transient in the SBWR-200 and SBWR-1200 reactors through code modeling and integral system testing. The focus of the research was the analysis of a major design base accident (DBA) such as a LOCA. The RELAP5/MOD3 best estimate reactor thermalhydraulic code was used and its applicability to the SBWR reactor safety system evaluation was examined. An anticipated transient with scram was simulated using RELAP5 code for SBWR-200. The transient considered was accidental closure of the main steam isolation valve (MSIV). The integral tests were performed to assess the safety systems and responses of the emergency core cooling systems to various loss of coolant accidents in a scaled facility called PUMA (Purdue University Multi-Dimensional Integral Test Assembly).

The performance of the safety systems of the SBWR-200 and SBWR-1200 reactors with failure of some safety systems was also studied. The focus of the research was the analysis of a beyond major DBA such as a LOCA. Integral tests for large break LOCA in the SBWR-200 and SBWR-1200 were performed each with one of three PCCS condenser unavailable during the accident. The following sections describe these efforts and the progress achieved in the safety study of the SBWR-200 and SBWR-1200 reactors.

6.1 PUMA Facility

The main goals of the experiments were to demonstrate the safety systems, validate the code applicability using a scale-up process, and evaluate the safety code for safety analysis. The performance of the safety system of the SBWR, the automatic depressurization systems (ADS), gravity driven cooling system (GDCS), passive containment cooling system (PCCS) and suppression pool (SP) are evaluated for various break size LOCA scenarios. The breaks considered were small break and large break and are described in the following sections.

The integral tests were performed on PUMA facility. The PUMA test facility at Purdue University is the only operational facility in the country that models next generation SBWR-relevant systems and components to produce data for SBWR applications. The PUMA facility was designed based on the three level scaling methodology described in the chapter 4 on scaling. The PUMA facility has a scale of $\frac{1}{4}$ in height, $\frac{1}{400}$ in volume, and a time scale of $\frac{1}{2}$ [6.1] with

respect to a midsize GE SBWR. The power is scaled by 1/200 of the prototype and the pressure is scaled 1:1. The facility contains all the important safety and non-safety systems of the SBWR that are pertinent to the postulated loss of coolant accident transients. PUMA has 400 instruments to provide phenomena understanding, quantification, evaluation, and flow visualization. The measured data are used for model development and for assessment of the RELAP5 code. The reactor pressure vessel has detailed internal components such as the core with fuel and by-pass channels, radial power distribution, lower plenum, downcomer, chimney and separator. The reactor core containing heater rods is designed with radial power distribution and bypass channels. Details of the instrumentation and PUMA components are available in a NUREG report [6.1]. The relative sizes of the piping and break sizes are based on the PUMA scale. The scaling criteria described in the PUMA design report [6.1] provide a guideline for the size of the break, pipe losses, vessel sizes, elevations, etc. Using the scaling methodology, the measured data can be easily translated to the prototype reactor configuration and time scale. PUMA is the only facility in the world that is well scaled and that has all safety-related components of the SBWR.

A new reactor pressure vessel made of 304 stainless steel was constructed to replace the old vessel, which had developed leaks. The new vessel was installed and has been state certified for operation. In Figure 6.1 shows the PUMA RPV design with internals. Figure 6.2 is the PUMA facility schematic. Integral tests were performed based on well-established test procedures and the quality assurance program. The details on the test procedure and the PUMA Q/A program are available in documents [6.2-6.3].

6.2 PUMA Integral Tests For SBWR-200 Safety Analysis

Two types of loss of coolant accidents (LOCA) were performed using the PUMA integral test facilities. These LOCAs are main steam line break (MSLB) and bottom drain line break (BDLB). For each accident case, two sets of successful test data have been obtained according to the established PUMA integral test procedures. The PUMA facility can handle transient of the accident from 1.03 MPa (150 psia) pressure to low pressures. Hence initial conditions at 1.03 MPa pressure are required to conduct integral tests. RELAP5/MOD3 code was used to obtain initial conditions for integral tests. A RELAP5 model of the SBWR-200 was prepared and the main steam line break (MSLB) and bottom drain line break (BDLB) was simulated. For this

simulation, the reactor was initially assumed to be at full power and normal operation condition and then the break was initiated. The scenario followed the progression of the accident including reactor scram and the ADS actuation. From this simulation, system thermal-hydraulic conditions at 1.03 MPa were obtained for the PUMA integral tests. It is widely accepted in the community that the early blowdown phase can be well predicted by the current safety analysis codes. Thus the system conditions at 1.03 MPa calculated by the code can be used as the test initial conditions. All the important characteristics of passive safety systems such as the ECCS and long term cooling appear below this pressure.

6.2.1 Main Steam Line Break LOCA

MSLB is a well-known design basis accident to investigate the SBWR-200 safety performance under the abnormal operation condition. It is a large break LOCA accident that requires ADS discharging and fully functioning of safety systems. Since one of the innovative SBWR-200 safety designs is to transfer most of the design base accidents into a large break LOCA via ADS discharge, MSLB is a particular representative case for the SBWR-200 design safety analysis. The PUMA MSLB test schematic is shown in Figure 6.3.

PUMA MSLB tests last 8 hours, which corresponds to 16 hours in the plant. After the break on the main steam line B was initiated, coolant inside the RPV quickly flashed and flowed out through the break site. Critical flow was quickly achieved at break site and the pressure on the RPV side dropped rapidly under the control of the ADS. This RPV depressurization allowed the injection of the GDCS. The MSLB blow-down phase lasted between 200 and 300 seconds. Following the GDCS injection, highly subcooled GDCS water suppressed the boiling in the RPV. The GDCS drain time lasted for about 3500 seconds. The PCCS started working after the GDCS injection was terminated. The RPV started boiling again due to continued decay heat. The reactor primary system and containment reached a relatively stable pressure level for the remaining period of the test. In this phase no significant change in the system temperature and pressure was observed during the long-term cooling phase.

6.2.2 Bottom Drain Line Break LOCA

BDLB accident is a design-basis small break LOCA in the SBWR. The PUMA BDLB test simulated the accident scenario in the SBWR by introducing the break located at the junction between the bottom drain line (BDL) and the reactor water clean up line (RWCU). The lowest

break elevation is considered to be the most serious design base accident in the SBWR design. The PUMA BDLB test schematic is shown in Figure 6.4.

PUMA BDLB test lasted 16 hours. Similar to MSLB, the PUMA BDLB test progression also showed three phases. The ADS actuated immediately after the break initiation. The critical flow occurred and RPV water level dropped below RWCU level. The blow-down phase lasted for approximately 100 seconds. At the end of the blow-down phase GDCS water started to drain after the drywell and RPV pressures equalized and it lasted up to 2750 seconds. The PCCS and ICS started working immediately after the GDCS injection terminated. After the SP/RPV equalization line opened, the system pressure was stabilized and the long-term cooling phase was achieved.

6.2.3 General Observation on MSLB and BDLB LOCAs

Both the MSLB test and BDLB test conducted on PUMA clearly indicated the three-phase LOCA accident progression, namely, the blow-down phase; the GDCS drain phase and the long-term cooling phase. These three accident phases have different dominant phenomena and characteristic times so that they can be analyzed separately. The details of these LOCA and safety system behavior are presented in the following sections.

The PUMA integral tests demonstrated that all of the SBWR engineered safety systems worked as designed. Furthermore, the transient behaviors following the break were relatively simple and clear, with three main phases. Each phase was clearly identifiable and the phenomena involved can be characterized by the instrumentation installed in the PUMA facility. The integral tests also indicated that PUMA was well designed in terms of integral test performance and control.

6.3 PUMA Integral Test Simulation Using RELAP5 for SBWR-200

The post- PUMA integral test simulations were carried out using the reactor safety code RELAP5. The integral test simulations examined preliminary code applicability at the scaled facility level (PUMA) as well as SBWR-200 key safety system performance. The RELAP5 nodalization diagram for the PUMA facility is shown in Figure 6.5. This PUMA system model includes 200+ control volumes and junctions and 100+ heat structures. It was separated into several distinct sections according to the function, i.e. the primary system, containment, and safety systems such as GDCS, ICS and PCCS.

The RPV is the most critical part in the PUMA system modeling. It is carefully modeled in order to preserve the RPV water inventory. The fuels in the core are grouped into two rings (outer/inner) and divided into several parallel channels with cross-flow junctions. A simple separator model is used with the separator tubes, modeled as vertical standpipes. PUMA containment is composed of the DW and the SC. They are simply modeled as large volumes without heat sources. A one-dimensional approach is applied on the various PUMA safety components modeling, such as GDCS, PCCS and ICS. Pipelines such as the SP equalization line and various supply lines and drain lines are modeled by using RELAP5's pipe components.

Three types of heat slabs are used in this model. The heat structures adjacent to the core channel contain heat sources. The heat generation rate is determined from the SBWR decay power curve as stated in the SSAR report [6.1]. The heat structure adjacent to the PCCS and ICS condenser tubes is modeled as the heat sink. The third type of heat structure modeled is the heat loss from the vessel wall.

Volume and junction options can be very important for the transient calculation. For the volume control options, mixture level tracking is turned off in all components since otherwise an unaffordable computational speed reduction will be encountered. The water packing and rod bundle options are generally turned off and the vertical stratification, wall friction and non-equilibrium models are turned on. The default for all the junctions is to turn off the choking option except for the junctions on the break lines and ADS lines. The CCFL option and abrupt area option are not used.

The desire of full-length test simulation requires a carefully designed problem time control. A semi-implicit method with the mass error estimate method is used to control the problem time step selection. The main computation limitation to the time step is the Courant limit of critical flow in the blow-down phase. For this reason, the break is modeled in a relatively simple way.

The main steam line transfers steam from the RPV steam dome to the steam turbine. The rupture of one of the main steam lines i.e., MSLB, results in a rapid system depressurization and coolant loss. Pressure losses in the broken main steam line, including friction loss and form loss, were carefully calculated and were confirmed by the characteristic test results. The standard critical flow options are latched on for the break junctions. The same test break geometry and orifice sizes as in the PUMA facility were used in the RELAP5 calculation.

According to the SBWR design, the bottom drain line is a small diameter pipe connected to the RPV bottom and to the RWCU line. The RWCU is connected to the middle of the RPV. The functions of the BDL and RWCU are to clean up the coolant system and to generate RPV internal circulation. The BDLB is triggered by the break of the connection between the BDL and RWCU resulting in flow out of the RPV from two paths. The break on BDL was modeled as an orifice critical flow with critical flow option reset while the break on RWCU line was modeled as a pipe critical flow and the critical flow option was latched on. The code provided discharge coefficients for both of these orifices.

6.4 Code Applicability Analysis for SBWR-200

The RELAP5 code applicability to the small and large break LOCA scenario was examined by comparing code predictions with the PUMA test data. By proper scale-up process, the overall PUMA system responses and safety behaviors can be extended to the SBWR-200 safety analysis. There are three important SBWR safety issues, viz., the primary system water inventory, the containment integrity and the engineered safety system behaviors. In this section, the comparisons between the RELAP5 simulation predictions and the counterpart integral test data are shown with respect to these key issues.

6.4.1 PUMA MSLB Test Prediction

PUMA MSLB transient calculation was carried out based on the system initial conditions at 1.03 MPa. A long-term MSLB transient calculation, up to 30,000 s, was performed and the run-time ratio of 3.34 was obtained. In the following section key representative test results are presented

6.4.1.1 Downcomer Collapsed Water Level

The downcomer collapsed water level, which is the measure of the RPV water inventory is shown in Figure 6.6. The code predicted downcomer collapsed water level agrees with the test data for all three accident phases reasonably well. In the early blow down stage of the transient, the downcomer collapsed water level drops due to the break flow out of the RPV. It then increases due to the GDCS injection. Both the test and the code prediction of the water level show consistent trends in the blow down stage. The minimum level in the figure corresponds to the starting time of the GDCS injection. It was higher than the top of the active fuel (TAF). Since the minimum downcomer collapsed water level was well above the TAF, the core remains

covered during the entire transient. This result suggests that the SBWR design is safe from core uncover for MSLB accident. The agreement of the downcomer water level also shows the primary system mass inventory can be fairly modeled by REALP5. The general strategy of the SBWR safety is to avoid core uncover for the design base accidents. This should significantly increase the probability of avoiding the core damage.

6.4.1.2 Drywell Pressure

The drywell and suppression chamber make up the SBWR containment boundary. The containment integrity is secure whenever pressure inside DW and SC is below the design limit. In most period of the transient, the SP pressure follows the DW. The DW pressure trace is shown in Figure 6.6. According to MSLB test analysis, the DW pressure rises quickly at the initial blow down stage and equalizes with the RPV pressure. It then decreases because of the GDCS water injection into the core which collapses steam void in the core. The pressure increases again after the GDCS flow terminates and it stabilizes during the long-term cooling phase. The code predicted DW pressure shows similar trend as the test data, confirming that the containment integrity is maintained for the entire MSLB transient. However, the predicted DW pressure is higher than the test data in the long-term cooling phase. This discrepancy is mainly due to the poor RELAP5 SP stratification model and the inadequate PCCS modeling. Though the RELAP5 containment pressure prediction is conservative, it may be necessary to develop two-dimensional model for detail containment phenomena analysis.

6.4.1.3 GDCS Drain Flow

GDCS is designed as the main emergency core cooling system for the SBWR-200. The GDCS drain flow rate indicates the GDCS behavior in the MSLB. Figure 6.7 shows that the code predicted GDCS flow rate agreed well with the test data. The GDCS starts to drain after the primary system pressure is equalized with the containment pressure. The flow rate change indicates the gravity head change during the drain process. The small flow rate after the main GDCS flow termination is due to the returning flow from the PCCS condensation line. This indicates that the GDCS functions properly as designed. It also indicates that the GDCS drain process is well represented by RELAP5.

6.4.1.4 Decay Heat removal

PCCS and ICS are two SBWR safety components primarily responsible for decay heat removal in the transients. The PCCS and ICS heat removal rate are plotted with the decay power

curve in Figure 6.8. The predicted heat removal rates show that the ICS works primarily on the early blow-down stage while the PCCS functions at the later stage of the accident. The prediction of ICS performance confirms the test data. The comparison also shows that PCCS functions automatically after re-boiling of the RPV water, revealing the passive PCCS performance as designed. The PCCS heat removal rate shows the correct trend since it follows the power decay trend. The code calculated PCCS heat removal rate approaches the decay power input at the later transient, indicating PCCS is the primary heat removal component for SBWR accidents. In the long term cooling phase.

6.4.2 PUMA BDLB Test Prediction

PUMA BDLB calculation simulates 58,500 seconds transient performance. The CPU to problem run time of 8.9 was achieved. The assessment of the RELAP5 code for the BDLB transient prediction was performed with the key parameter comparison similar to the MSLB case.

6.4.2.1 Downcomer Collapsed Water Level

The downcomer-collapsed water level is shown in Figure 6.9. BDLB is a small break LOCA; however, it is potentially the most challenging LOCA compared to other the larger break LOCAs such as MSLB. Illustrated in the blow-down stage of the transient, the downcomer collapsed water level dropped due to the break flow through the RWCU line. It then increased due to the GDCS water injection into the RPV. The minimum level in the figure is slightly lower than the TAF, indicates a possibility of core uncover for a short period in BDLB LOCA. However, this is unlikely because the two-phase level is higher than the collapsed water level. The result also implies that the break elevation but not the break size is the dominant safety factor in the SBWR reactor. The code predicted downcomer collapsed water level agrees with the test data for all three accident phases reasonably good.

6.4.2.2 Drywell Pressure

The BDLB DW pressure modeling result is not as good as the level prediction, though it still preserves the general trends (Figure 6.10). The DW pressure as indicated by both code prediction and test data is well below the design limits confirming that the containment remains secure for the entire BDLB transient. However, the code under-predicts the pressure in the GDCS drain phase and over-predicts the pressure in the long-term cooling phase. Though the prediction of

the containment pressure in the BDLB case is conservative, it is necessary to improve RELAP5 models for accurate containment phenomena analysis.

6.4.2.3 GDCS Drain Flow

The GDCS drain flow rate calculated by PUMA BDLB prediction is shown in Figure 6.11. The predicted GDCS behavior shows good agreement with the test data. The GDCS starts to drain after the primary system pressure equalized with the containment pressure. The flow rate changes according to the gravity head. This indicates that the GDCS functions properly in BDLB case as designed. It also indicates that the GDCS drain process is well represented by RELAP5.

6.4.2.4 Decay Heat Removal

PCCS and ICS are two SBWR safety components primarily responsible for decay heat removal in the transients. The SP functions as the primary heat sink for the initial RPV energy inventory. The PCCS and ICS heat removal rate are plotted with the decay power curve in Figure 6.12. The predicted heat removal rate shows that the ICS works primarily during early blow-down stage while the PCCS functions at the later stage of accident. The prediction of ICS performance confirms the test data. The comparison also shows that PCCS functions automatically after the RPV water is reboiled, revealing the PCCS performance as designed. The PCCS heat removal rate follows the power decay trend. The code calculated PCCS heat removal rate approaches the decay power input at the later transient, indicating PCCS is the primary heat removal component for SBWR accidents during the long term cooling phase.

6.5 MSIV Closure Transient for SBWR-200

An anticipated transient with scram was simulated using RELAP5/MOD3 code for the SBWR-200 reactor. The transient problem considered was the accidental closure of the main steam isolation valve during normal operation of the reactor followed by scram. This transient is similar to the station blackout with scram. The simulation was carried out for 16 hours following the accident. Figure 6.13 shows the predicted RPV pressure. The RPV pressure increases from the normal operation pressure of 6.9 MPa (1040 psia) to about 8 MPa immediately following the closure of the MSIV. The reactor is scrammed during this period. Then the pressure falls to slightly above normal operational pressure until 500 seconds following the accident, due to the opening of the SRVs. The pressure recovers slightly following the closure of the SRVs. The ICS condensers are also active and the pressure starts falling after 1000 seconds due to steam

condensation in ICS. At 16 hours the pressure falls asymptotically to 0.5 MPa. In Figures 6.14 the RPV downcomer collapsed water level is shown. During the first 40 seconds the water level decreases as the steam is vented from the SRVs and the ICS is condensing the steam. After 100 seconds, the collapsed water level decreases as the void in the RPV is decreased. It can be seen from this figure that the ADS is never actuated for this transient as the Level 1 (10.6m) for ADS actuation is far below the RPV water level (13.2m) at 16 hours. Initially when the RPV pressure is high, the ICS condensation is high. The ICS is active during the entire 16 hour period and its condensation rate depends on the RPV pressure. As the RPV pressure decreases the condensation also decreases. In Figure 6.15 the heat removal rate from the ICS is shown and is compared with the decay heat produced in the RPV core. The ICS is very effective in removing the RPV decay heat. After 40,000 seconds the ICS heat removal rate is larger than the decay heat, indicating the effectiveness of the ICS in handling this accident.

In summary, for the SBWR-200 MSIV closure accident, the minimum RPV water level reached is 13.2 m compared to the TAF of 6.5 m. The ADS is not actuated and no ECC is required. The RPV core is adequately cooled by ICS. The ICS is quite effective in removing decay heat. The long term (16 hours) RPV pressure is steady at 500 kPa.

6.6 PUMA Integral Tests For SBWR-1200 Safety Analysis

Two types of loss of coolant accidents (LOCA) were studied for SBWR-1200 similar to SBWR-200. The details of the testing and performance of the safety system are similar to the SBWR-200 presented in section 6.2 and hence are not repeated here.

6.7 SBWR-1200 Test Simulation Using RELAP5

Following the integral tests, the safety system performances of SBWR-1200 were simulated using the reactor safety code RELAP5. This system model includes 300+ control volumes, 400+ junctions and 200+ heat structures. The whole system was separated into several distinct sections according to their functions, i.e. the primary system, containment, and safety systems such as GDSCS, ICS, and PCCS. The one-dimensional approach was applied. The multi-component

systems were modeled individually, such as PCCS, ICS, and GDCS systems. The other simulation details are similar to the SBWR-200 given in Section 6.3.

6.8 Code Applicability Analysis for SBWR-1200

The RELAP5 predictions for the small and large break LOCA scenario were examined by comparing code predictions with the scaled SBWR-1200 test data obtained from the PUMA facility. The results are similar to the SBWR-200 case.

6.8.1 SBWR-1200 MSLB Test Prediction

In Figure 6.16 the scaled up PUMA test data of the RPV pressure is compared with the RELAP5 prediction. The comparison shows very good agreement between the code prediction and the experimental data. The DW pressure trace is shown in Figure 6.17. The code predicted DW pressure shows a similar trend as the test data, confirming that the containment integrity is maintained for the entire MSLB transient. However, the predicted DW pressure is higher than the test data from 500 seconds to 4200 seconds. This discrepancy is mainly due to the poor RELAP5 SP stratification model and the inadequate PCCS modeling. The downcomer collapsed water level, which is the measure of the RPV water inventory is shown in Figure 6.18. The code predicted downcomer collapsed water level agrees with the test data for all three accident phases reasonably well. The PCCS and ICS heat removal rate from PUMA test data are plotted with the decay power curve in Figure 6.19. The RELAP5 predicted decay heat, and PCCS and ICS heat removal rates are shown in Figure 6.20. Both the experimental and predicted heat removal rates show that the ICS works primarily in the early blow-down stage and contributes a lower heat removal rate compared to the PCCS at later stage. The PCCS functions at the later stages of the accident.

6.8.2 SBWR-1200 BDLB Test Prediction

In Figure 6.21 the scaled up PUMA test data of the RPV pressure is compared with the RELAP5 prediction. The comparison shows a very good agreement between the code prediction and the experimental data. The blowdown lasts about 200 seconds. The RPV pressure reaches a steady pressure of 250 kPa at 32 hours following the accident. The DW pressure comparison is shown in Figure 6.22. The predicted DW pressure shows a similar trend as shown in the experimental data. The downcomer-collapsed water level is shown in Figure 6.23. The code

predicted downcomer collapsed water level agrees with the test data for all three accident phases. The test data of the PCCS and the ICS heat removal rate are plotted with the decay power curve in Figure 6.24. The predicted heat removal rates of PCCS and ICS and the decay heat curve are shown in Figure 6.25.

6.9 PUMA Integral Tests for Beyond DBA

Integral tests were conducted for beyond design basis accident. Loss of coolant accident (LOCA) have been studied with failure of one of three PCCS condenser using the PUMA integral test facility during the third year. The LOCA considered was the main steam line break (MSLB). For the accident case, one set of successful test data has been obtained according to the established PUMA integral test procedures..

The 8 hour MSLB test conducted on the PUMA clearly indicated the three-phase LOCA accident progression, namely, the blow-down phase; the GDCS drain phase and the long-term cooling phase. These three accident phases have different dominant phenomena and characteristic times so that they can be analyzed separately. The transient behavior for SBWR-1200 and SBWR-200 are similar during MSLB with PCA closed off. Hence the discussion on the figures is given for SBWR-1200 reactor only.

6.9.1 Safety Analysis for SBWR-1200 LOCA with PCCS Failure

In Figure 6.26 the RPV pressure is shown. The blowdown lasts about 350 seconds. The RPV pressure reaches a steady pressure of about 270 kPa at 16 hours following the accident. The DW pressure trace is shown in Figure 6.27. During a large portion of the transient, the SP pressure follows the DW. The downcomer collapsed water level, which is the measure of the RPV water inventory, is shown in Figure 6.28. This result suggests that the SBWR design is safe from core uncover for the MSLB accident. The general strategy of the SBWR safety analysis is to avoid core uncover for the design base accidents. This should significantly increase the probability of avoiding the core damage.

The PCA condenser unit was closed off in this test. PCB and PCC condenser units were operational. In Figures 6.29 and 6.30 the steam inlet flow rate to the PCB and PCC units are shown. Both the condenser units show similar heat removal capacity. Overall condensation from ICS is smaller than the PCCS.

6.9.2 Safety Analysis for SBWR-200 LOCA with PCCS Failure

The corresponding figures for the SBWR-200 against the subtitle are listed below. The behavior of the transient phenomena is similar to the SBWR-1200 reactor.

RPV Pressure -Figure 6.31

Drywell Pressure-Figure 6.32

Downcomer Collapsed Water Level - Figure 6.33.

PCCS Performance- Figures 6.34, 6.35.

6.9.3 *Conclusions on Safety Study for LOCA with PCCS Failure*

The reactors SBWR-1200 and SBWR-200 behave very similar during main steam line break with one PCCS condenser non-operational. The water level in the RPV is above the top of the active fuel during the entire transient for both reactors. The emergency core cooling system, the GDCCS functions properly during this accident. The unavailability of one of the PCCS condensers is minimum on the overall safety of the reactor. The two remaining condenser units adequately condense steam in the containment and maintain the drywell pressure below design pressure.

6.10 References

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Figure 6.1 PUMA reactor pressure vessel and internal

Figure 6.2 Overall schematic of PUMA Facility

Detailed Break Line

Figure 6.3 PUMA main steam line break schematic

Figure 6.4 PUMA Bottom Drain Line Break Schematic

Figure 6.5 PUMA MSLB for SBWR-200, downcomer collapsed water level

Figure 6.6 PUMA MSLB for SBWR-200, drywell pressure

Figure 6.7 PUMA MSLB for SBWR-200, GDCS- A drain injection flow rate

Figure 6.8 PUMA MSLB for SBWR-200, decay power removal

Figure 6.9 PUMA BDLB for SBWR-200, downcomer collapsed water level

Figure 6.10 PUMA BDLB for SBWR-200, drywell pressure

Figure 6.11 PUMA BDLB for SBWR-200, GDCS- A drain injection flow rate

Figure 6.12 PUMA BDLB for SBWR-200, decay power removal

Figure 6.13 RELAP5 simulation for SBWR-200 MSIV closure, RPV pressure

Figure 6.14 RELAP5 simulation for SBWR-200 MSIV closure, RPV downcomer collapsed water level

Figure 6.15 RELAP5 simulation for SBWR-200 MSIV closure, ICS heat removal rate

Figure 6.16 Comparison of RELAP5 prediction and scaled-up PUMA MSLB test for SBWR-1200, RPV pressure

Figure 6.17 Comparison of RELAP5 prediction and scaled-up PUMA MSLB test for SBWR-1200, drywell pressure

Figure 6.18 Comparison of RELAP5 prediction and scaled-up PUMA MSLB test for SBWR-1200, downcomer collapsed water level

Figure 6.19 Comparison of RELAP5 prediction and scaled-up PUMA MSLB test for SBWR-1200, decay heat removal, scaled up test data

Figure 6.20 Comparison of RELAP5 prediction and scaled-up PUMA MSLB test for SBWR-1200, decay heat removal, code predictions.

Figure 6.21 Comparison of RELAP5 prediction and scaled-up PUMA BDLB test for SBWR-1200, RPV pressure

Figure 6.22 Comparison of RELAP5 prediction and scaled-up PUMA BDLB test for SBWR-1200, drywell pressure

Figure 6.23 Comparison of RELAP5 prediction and scaled-up PUMA BDLB test for SBWR-1200, downcomer collapsed water level

Figure 6.24 Comparison of RELAP5 prediction and scaled-up PUMA BDLB test for SBWR-1200, decay heat removal, scaled up test data

Figure 6.25 Comparison of RELAP5 prediction and scaled-up PUMA BDLB test for SBWR-1200, decay heat removal, code predictions

Figure 5.26 RPV steam dome pressure, MSLB test for SBWR-1200 with PCA Condenser Off

Figure 6.27 Upper drywell pressure, MSLB test for SBWR-1200 with PCA Condenser Off

Figure 6.28 RPV downcomer collapse water level, MSLB test for SBWR-1200 with PCA Condenser Off

Figure 6.29 Steam inlet flow rate to PCB unit, MSLB test for SBWR-1200 with PCA Condenser Off

Figure 6.30 Steam inlet flow rate to PCC unit, MSLB test for SBWR-1200 with PCA Condenser Off

Figure 6.31 RPV steam dome pressure, MSLB test for SBWR-200 with PCA Condenser Off

Figure 6.32 Drywell pressure, MSLB test for SBWR-200 with PCA Condenser Off

Figure 6.33 RPV downcomer collapse water level, MSLB test for SBWR-200 with PCA Condenser Off

Figure 6.34 Steam inlet flow rate to PCB unit, MSLB test for SBWR-200 with PCA Condenser Off

Figure 6.35 Steam inlet flow rate to PCC unit, MSLB test for SBWR-200 with PCA Condenser Off

7. SBWR NEUTRONICS DESIGN AND ANALYSIS

7.1 Introduction

The following section summarizes the research performed as part of the neutronics design and analysis of a 200 and 1200 MWe SBWR. The first section reviews the analysis methods and their validation using the 600 MWe SBWR. The next sections discuss the detailed design and analysis of the 200 and 1200 MWe SBWRs.

7.2. Analysis of the 600 MWe SBWR Core

The neutronics and thermal-hydraulics analysis of an advanced reactor such as the SBWR requires the use of three basic types of codes; a lattice physics code to generate few group cross sections, a neutronics core simulator to calculate the core eigenvalue coupled to a depletion code to perform the core depletion calculations, and a thermal-hydraulics code to provide temperature and fluid feedback to the core simulator. The three codes used for the SBWR analysis will be described in the following section, which will also describe their application to the GE 600 MWe design.

7.2.1 *Purdue Reactor Core Analysis Code System*

The Purdue Reactor Core Analysis Code System is depicted in Figure 7.1 and consists of the HELIOS [2] lattice physics code, the U.S. NRC core neutronics simulator PARCS [4], and the system thermal-hydraulics codes RELAP5 [7] or TRAC-M [14]. The coupling of RELAP5 and TRAC-M to PARCS is performed using the message passing interface PVM, and has been described Reference 5. A separate depletion module, DEPLETOR [6], was developed to perform core burnup calculations and is coupled to PARCS using the same PVM message passing interface.

The PARCS code (Purdue Advanced Reactor Core Simulator) determines the neutron flux distribution by solving the neutron diffusion equation [4]. The code has the capability to calculate a three dimensional flux solution for Cartesian and non-orthogonal geometries in multiple energy groups. In the work here, PARCS was coupled with the thermal-hydraulics code RELAP5 to provide the coupled neutronics/thermal-hydraulics solution [5].

Figure 7.1 Purdue Reactor Core Analysis Code System (PARCS)

RELAP5 [7] is a well established thermal-hydraulics code that solves the nonequilibrium, inhomogeneous two-phase flow problem using the six-equation, two fluid model. This code is capable of simulating full system models and has been extensively benchmarked for most light water reactor applications. However, for the work here it was necessary to make some modifications to the RELAP5 for boiling water reactor applications. Specifically, the Hench-Gillis critical heat flux (CHF) model was implemented in RELAP5 for predicting dryout in a SBWR fuel assembly. The following section will describe the application of the HELIOS/PARCS/RELAP5 code system to the GE 600 MWe SBWR design, to include results from applying the Hench-Gillis CHF model in RELAP5 to the GE 600 MWe SBWR.

7.2.2 Code Qualification using the 600 MWe SBWR

The original GE design of the 600 MWe SBWR was used as a basis for the 200 MWe SBWR design. In order to properly verify the tools that were used in the design of the 200 MWe SBWR, a model was first built of the 600 MWe SBWR and comparisons were made to the reactor performance as reported by GE. Some details of the GE 600 MWe design such as the burnable loading in the fuel assembly were not available in the open literature, and therefore the results reported are not expected to reproduce exactly the GE results as reported in the SSAR.

An equilibrium core fuel loading was developed using the SBWR fuel assembly designs. The overall core parameters for the 600 MWe core are given in Table 7.1.

Table 7.1 Design Parameters for the 600 MWe Core

Core Property	600 MWe SBWR
Number of fuel assemblies	732
Fuel assembly arrangement	30X30
Core diameter (m)	4.730
Active fuel length (m)	2.743
Length of gas plenum (m)	0.3048
Water rods (total)	2220
Control Blades	177

The core loading pattern used for the three-batch, equilibrium cycle core is quarter core rotationally symmetric and is shown in Figure 7.2. A control cell loading concept was used to minimize the impact of control rod movement on the radial power profile [9].

Figure 7.2 Fuel loading pattern for the 600 MWe SBWR

A RELAP5 thermal-hydraulics model was developed with the nodalization shown in Figure 7.3. The same model with some modifications was used for both the 200 and 1200 MWe core.

Figure 7.3 RELAP5 nodalization diagram of the SBWR core

A comparison of the calculated thermal-hydraulics data for the Beginning of Cycle (BOC) state of the GE 600 MWe core design and the Purdue 600 MWe design are shown

in Table 7.2. Overall, the thermal-hydraulics parameters match the GE results fairly well. The core flow rates and core averaged pressures match almost exactly, whereas there are some differences in the void fractions and flow qualities.

Table 7.2. Comparison of GE and Purdue 600 MWe SBWR TH Properties at BOC

	GE SBWR	Purdue SBWR
Power (MWth)	2000	2000
Core Flow Rate (kg/hr)	27.2X10 ⁶	27.3X10 ⁶
Power Density (kW/liter)	41.5	41.5
MCPR (minimum)	1.32	1.34
Average Core Void Fraction	0.434	0.468
Maximum Exit Void Fraction	0.836	0.793
Core Pressure (MPa)	7.239	7.239
Core Pressure Drop (kPa)	48.0	43.3
Average Exit Quality	0.143	0.170

The core was then depleted and the results are shown in Table 7.3. The control rod notches represent the number of control rod levels that are inserted into the reactor core at each burnup state.

The cycle burnup of about 14.1 GWd/MT predicted here represents a cycle length of 1.95 years which is similar to the 2 year (14.3 GWd/MT) cycle length predicted by GE. It should be noted that the minimum critical power ratio predicted for the cycle is 1.33 and occurs at the beginning of the cycle where the maximum radial power peaking occurs. The minimum steady state critical power ratio reported by GE was 1.32 [1]. The results of the 600 MWe SBWR analysis provided confidence in the accuracy of the Purdue Reactor Core Analysis Code System for SBWR analysis. A 200 MWe SBWR core was then developed based on the 600 MWe design.

7.3 Design and Analysis of the 200 MWe SBWR

The 200 MWe design was based on the 600 MWe but scaled to accommodate the smaller size of the core. All of the passive safety system features contained in the 600

MWe design were retained for the 200 MWe core. The following sections will first describe the design of the fuel assemblies and core, and then present results of core fuel cycle analysis.

Table 7.3 Fuel Cycle Depletion Results for 600 MWe SBWR

Burnup⁽¹⁾ (GWd/T)	K_{eff}	Control Rod Notches	Maximum Power Peak	Radial Power Peak	Axial Power Peak	MCPR
0.0	1.00023	514	1.874	1.335	1.450 (5) ⁽²⁾	1.33
1.2	1.00090	498	1.790	1.301	1.363 (5)	1.38
2.3	1.00089	459	1.756	1.292	1.322 (5)	1.40
3.5	0.99900	545	1.795	1.270	1.329 (5)	1.45
4.7	0.99913	545	1.797	1.273	1.307 (5)	1.43
5.9	0.99984	545	1.817	1.278	1.297 (5)	1.42
7.0	1.00065	506	1.885	1.278	1.311 (5)	1.41
8.2	0.99912	585	2.056	1.276	1.396 (5)	1.46
9.4	0.99965	528	2.096	1.299	1.397 (5)	1.41
10.6	0.99965	497	2.031	1.316	1.345 (5)	1.39
11.7	0.99996	418	1.869	1.319	1.254 (5)	1.38
12.9	0.99988	262	1.812	1.306	1.398 (8)	1.40
14.1	0.99318	0	2.158	1.277	1.448 (4)	1.40

¹The heavy metal loading here was normalized to the 600 MWe for a consistent comparison

²Indicates the axial level at which the maximum axial peak occurs (1 = core entrance, 20 = core exit)

7.3.1 200 MWe SBWR Core Design

The primary design constraint for the 200 MWe core was to maintain a cycle length and safety performance similar to the 600 MWe core. The principal design changes to the fuel assembly were to increase the fuel enrichment and burnable absorber loading. The major difference between the fuel assemblies used in the 600 MWe core and the fuel used in the 200 MWe core was the average fuel enrichment. Because the core leakage was larger in the smaller core, it was necessary to increase the enrichment from 3.95 wt% to

4.26 wt%. This increase in enrichment required changes in both the number and location of Gadolinia pins in the fuel assembly. The 8x8 fuel assembly design used for the 200 MWe SBWR is shown in Figure 7.4. The 8 burnable poison pins contained 1.8 w/o Gd and are located as shown below (BP pins are noted with multiple rings).

Figure 7.4 Fuel assembly design used for the 200 MWe SBWR

An equilibrium cycle was designed for the 200 MWe core. The principal core design parameters are shown in Table 7.4

Table 7.4 Design Parameters for the 200 MWe Core

Core Property	200 MWe SBWR
Number of fuel assemblies	256
Fuel assembly arrangement	18X18
Core diameter (m)	2.730
Active fuel length (m)	2.743
Length of gas plenum (m)	0.3048
Water rods (total)	750
Control Blades	60

A fuel loading pattern was developed for the 200 MWe core which was similar to the 600 MWe. However, the number of twice burned fuel was limited because of the smaller core size and increased core leakage which required a higher amount of reactivity to maintain the desired cycle length. The loading pattern for the 200 MWe core is shown in Figure 7.5.

Figure 7.5 Core loading pattern for the 200 MWe SBWR

Using the radial and axial assembly designs shown above, a RELAP5/PARCS model was constructed using one node per fuel assembly in the radial direction and 20 nodes per assembly in the axial direction. Some of the design parameters are shown in Table 7.5.

A comparison of the thermal-hydraulic performance between the GE 600 MWe core design and the Purdue 200 MWe design is shown in Table 6. The power density for the 200 MWe is slightly less than the GE design because the core power was scaled by 1/3 while the core size was reduced by only a factor of 0.35 to maintain a relatively circular core cross section. The reduction in power density means that the core flow rates did not scale with power. However, most of the remaining core averaged properties remained similar to the 600 MWe.

Table 7.5 Comparison of the 600 and 200 MWe Core TH Properties at BOC

	GE 600 Mwe	Purdue 200 MWe
Power (MWt)	2000	660
Core Flow Rate (kg/hr)	27.2×10^6	8.93×10^6
Power Density (kW/liter)	41.5	39.6
MCPR (minimum)	1.32	1.320
Average Core Void Fraction	0.434	0.455
Maximum Exit Void Fraction	0.836	0.844
Core Pressure (MPa)	7.239	7.243
Core Pressure Drop (kPa)	48	37
Average Exit Quality	0.143	0.183

The 200 MWe core was then depleted and the depletion results are shown in Table 7.6. A comparison with the results shown previously in Table 7.3 for the 600 MWe core indicate that the power peaking in the 200 MWe core is slightly higher and the minimum critical power ratio is slightly lower.

7.4 Design and Analysis of the 1200 MWe SBWR

The 1200 MWe design concept was based on the original 600 MWe SBWR as described in the SSAR. The system was scaled to accommodate the larger size of the core. The strategy behind the 1200 MWe design is to create a full size version of the original 600 MWe design. The larger reactor would be of more interest to the developed nations that have a large electricity demand. This design incorporates all of the passive safety system features contained in the 600 MWe design. Even though the size of the core increased, the fuel assemblies remained similar to the 600 MWe design. Instead of lowering the enrichment to account for the decreased leakage, the batch loading was varied, making use of more twice burned fuel. This increases the fuel utilization of the core, and decreases the amount of fresh fuel needed for each cycle. The design parameters and the fuel loading are shown in Table 7.7 and Figure 7.6.

Table 7.6 Fuel Cycle Depletion Results for the 200 MWe SBWR

Burnup⁽¹⁾ (GWd/MT)	K_{eff}	Control Rod Notches	Maximum Power Peak	Radial Power Peak	Axial Power Peak	MCP R
0.0	0.99994	224	2.051	1.256	1.519 (4) ⁽²⁾	1.46
1.2	1.00093	237	1.922	1.235	1.433 (5)	1.53
2.5	0.99980	224	1.925	1.242	1.416 (5)	1.52
3.7	1.00049	244	1.895	1.254	1.369 (5)	1.49
4.9	1.00073	204	1.905	1.277	1.354 (5)	1.44
6.1	1.00039	234	2.000	1.277	1.396 (5)	1.41
7.4	1.00032	242	2.154	1.284	1.461 (4)	1.36
8.6	1.00097	246	2.217	1.295	1.498 (4)	1.33
9.8	0.99963	239	2.158	1.303	1.490 (4)	1.32
11.1	0.99904	236	1.946	1.304	1.392 (5)	1.34
12.3	1.00060	197	1.727	1.306	1.258 (6)	1.33
13.5	0.99958	180	1.812	1.304	1.277 (13)	1.35
14.7	0.99915	131	2.124	1.316	1.45 (10)	1.34
14.9	1.00065	98	2.303	1.351	1.495 (9)	1.32

¹The heavy metal loading here was normalized to the 600 MWe for a consistent comparison

²Indicates the axial level at which the maximum axial peak occurs (1 = core entrance, 20 = core exit)

Table 7.7 Parameters for the 1200 MWe Core

Core Property	1200 MWe SBWR
Number of fuel assemblies	1464
Fuel assembly arrangement	x42y42
Core diameter (m)	6.690
Active fuel length (m)	2.743
Length of gas plenum (m)	0.3048
Water rods (total)	4440
Control Blades	350

Figure 7.6 1200 MWe core design

A comparison of the core averaged thermal-hydraulic properties between the GE 600 MWe core design and the Purdue 1200 MWe design is shown in Table 7.8 and the depletion results are shown in Table 7.9.

Table 7.8 Comparison of the 1200 MWe Core TH Properties, BOC

	GE SBWR	1200 MWe SBWR
Power (MWt)	2000	4000
Core Flow Rate (kg/hr)	27.2X10 ⁶	54.8X10 ⁶
Power Density (kW/liter)	41.5	41.5
MCPR (minimum)	1.61	1.266
Average Core Void Fraction	0.434	0.490
Maximum Exit Void Fraction	0.836	0.790
Core Pressure (MPa)	7.239	7.253
Core Pressure Drop (kPa)	48	37
Average Exit Quality	0.143	0.165

Table 7.9 Fuel Cycle Depletion Results for 1200 MWe SBWR

Burnup ⁽²⁾ (GWd/MT)	Control Rod Notches	Maximum Power Peak	Radial Power Peak	Axial Power Peak	MCPR
0.0	1214	2.396	1.318	1.630 (4) ⁽¹⁾	1.289
1.2	1187	2.260	1.300	1.528 (4)	1.318
2.3	1159	2.224	1.305	1.460 (4)	1.333
3.5	1257	2.341	1.328	1.465 (4)	1.363
4.7	1201	2.523	1.360	1.482 (4)	1.308
5.9	1270	2.657	1.403	1.529 (4)	1.266
7.0	1340	2.511	1.419	1.496 (4)	1.267
8.2	1201	2.308	1.413	1.435 (4)	1.290
9.4	1314	2.107	1.393	1.362 (5)	1.321
10.6	1196	1.889	1.363	1.283 (6)	1.377
11.7	1056	1.742	1.317	1.244 (10)	1.371
12.9	987	1.741	1.297	1.370 (15)	1.386
14.1	849	1.918	1.286	1.509 (14)	1.350
15.3	623	1.960	1.347	1.475 (12)	1.332

⁽¹⁾Indicates the axial level at which the maximum axial peak occurs (1 = core entrance, 20 = core exit)

⁽²⁾The heavy metal loading here was normalized to the 600 MWe for a consistent comparison

7.5 Summary of SBWR Designs

A summary of the SBWR designs is provided in Tables 7.10, 7.11 and 7.12

Table 7.10 Comparison of System Design Parameters

Components	GE SBWR 600 MW	SBWR 200 MW	Ratio (6/2)	SBWR 1200 MW	Ratio (6/12)
Core					
Number of fuel assemblies	732	256	3	1464	0.5
Fuel assembly arrangement	x30y30	x18y18		x42y42	
Core diameter (m)	4.730	2.730	1.7326	6.690	0.7070
Active fuel length (m)	2.743	2.743	1	2.743	1
Length of gas plenum (m)	0.3048	0.3048	1	0.3048	1
Water rods (total)	~2220	~750	2.96	~4440	0.5
Control Blades (total)	177	~60	2.95	~350	0.5057
Fuel pellet materials	UO ₂	UO ₂		UO ₂	
Vessel					
Hydraulic diameter (mm)	13.7	13.7	1	13.7	1
Total core flow area (m ²)	7.4	2.59	2.86	14.8	0.5
Core bypass flow area (m ²)	5.3	1.77	3.0	10.6	0.5
Core height (m)	3.830	3.830	1	3.830	1
Core shroud inner diameter (m)	5.150	3.050	1.689	7.210	0.714
Core shroud outer diameter (m)	5.250	3.150	1.667	7.310	0.718
Total vessel height (m)	24.612	24.612	1	24.612	1
Core downcomer width (m)	0.375	0.215	1.744	0.533	0.704
Core downcomer area (m ²)	6.63	2.21	3	13.26	0.5

Table 7.11 Comparison of TH Properties, BOC

	GE 600 MWe	600 MWe	200MWe	1200 MWe
Power (MWt)	2000	2000	660	4000
Core Flow Rate (kg/hr)	27.2X10 ⁶	27.3X10 ⁶	8.93X10 ⁶	54.8X10 ⁶
Power Density (kW/liter)	41.5	41.5	39.6	41.5
MCPR	1.61	1.337	1.320	1.266
Average Core Void Fraction	0.434	0.468	0.455	0.490
Maximum Exit Void Fraction	0.836	0.793	0.844	0.790
Core Pressure (MPa)	7.239	7.239	7.243	7.253
Core Pressure Drop (kPa)	48	43.3	37	37
Average Exit Quality	0.143	0.170	0.183	0.165

Table 7.12 Comparison of Fuel Cycle Calculations

	600 MWe	200MWe	1200 MWe
Cycle Length (GWd/MT)	14.1	14.9	>15.3
Cycle Length (yr)	1.95	2.1	>2.2
Maximum Radial Peak	1.335	1.316	1.419
Maximum Axial Peak	1.450	1.519	1.630
Maximum Power Peak	2.096	2.217	2.657

As indicated in the tables, the 600 MWe and 200 MWe show relatively similar power peaking values, whereas the 1200MWe design has a slightly higher power peaking. Nonetheless, it appears both the 200 MWe and 1200 MWe would be able to meet both the cycle length and safety margins. And in general, the results of the research here show that 200 and 1200 MWe SBWR cores can be designed with core performance parameters similar to the 600 MWe core.

7.6 References

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8. INSTABILITY ANALYSIS

The objectives under instability analysis are to 1) analyze the stability of the SBWR under different scenarios such as normal startup and abnormal startup, 2) design controls and orifices at the ore inlet for stable operation of the reactor and 3) develop design procedures to avoid instabilities. During this program, the simulation of the startup transients was preformed and instability characteristics were analyzed for SBWR-1200, SBWR-600 and SBWR-200 reactor designs and compared. In addition, program also assisted in developing procedure for simulating startup with PUMA, developed models with RAMONA-4B for these tests, and made preparation to simulate startup procedure with coupled neutronic-thermal-hydraulic calculations with RAMONA-4B code when actual procedure becomes available.

8.1 SBWR Instability

The Simplified Boiling Water Reactor (SBWR) is an advanced passive design using natural circulation for coolant flow without any active pumps. The startup procedure of the SBWR is of special importance since the low pressure; low flow and low power prevailing in the early part of the transient can become a precursor to instability. In a natural circulation system like the SBWR, the core flow is strongly coupled to the reactor core power. General Electric has proposed a procedure [8.1] for starting SBWR. Startup of the reactor begins with slow heat-up of the coolant in the reactor followed by removal of selected groups of control rods. Gradual increase of reactor power results in single-phase natural circulation flow within the reactor, while the system pressure is still very low. During this period, transition from a subcooled core to a saturated core with a subcooled chimney may be accompanied by creation of large vapor bubbles, which can initiate geysering type instability. Certain combination of pressure, flow and thermal-hydraulic conditions was required for geysering to occur in small-scale laboratory experiments such as Aritomi et al. [8.2,8.3] and Wang et al. [8.4]. According to these research findings, out of phase geysering in two parallel channels is limited to low-pressure and low-flow conditions. A subcooled chimney allows condensation of the large bubbles leading to instability in the flow within the channels. At higher power, loop type instability may follow the geysering instability, where the flow in the downcomer and the core oscillates in phase. Finally, density wave oscillations of higher frequencies can occur at higher power-to-flow conditions.

Although oscillations during startup have been demonstrated in the laboratory scale experiments, the two startups at the natural circulation Dodewaard reactor during 1992 have not shown any type of instability [8.1,8.5]. However, there are sufficient differences between the SBWR, Dodewaard reactor and the laboratory experiments, so that the conclusions made from these tests cannot be applied to the SBWR directly. Simulations of startup conditions for the SBWR (SBWR-600 and SBWR-200) have been performed at BNL using the RAMONA-4B code. This code uses 3D-neutron kinetics with a drift flux model for two-phase flow thermal-hydraulics. Since information on the spatial and temporal movements of the control rods are not available, in order to simulate the SBWR startup, a time-dependent power ramp profile has been imposed as a boundary condition and a thermal-hydraulic only calculation performed with four parallel coolant channels representing the reactor core. Thermal-hydraulic calculations over a long period of time (3~10 hours) are needed for a startup simulation. Simulations of the SBWR-600 startup were also performed by GE using the TRACG code [8.1,8.6].

8.2 Simulation of the SBWR Startup Transient

8.2.2 *Simulation of the SBWR Startup Using the RAMONA-4B Code*

8.2.2.1 The RAMONA-4B Code

The startup transients for SBWR-1200 have been simulated using the RAMONA-4B code developed at BNL [8.9,8.10]. The code models all the important components in the reactor pressure vessel (RPV), such as the reactor core, downcomer, lower plenum, upper plenum and riser, jet or internal pumps, steam separators and dryers, and the steam dome, as well as the control and plant protection systems, steam lines, balance of plant (BOP), and containment. SBWR specific components such as the isolation condenser (IC) and the standby liquid control system (SLCS) have also been modeled. The neutron kinetics is modeled with a time-dependent 3D diffusion theory with one and half group of prompt neutrons and six groups of delayed neutrons for a maximum of 804 neutronic channels each with a maximum of 24 axial nodes. Local thermal-hydraulic feedback is taken into account in terms of the changes in nodal two-group cross sections due to the local void, fuel temperature, and moderator temperature. The two-phase thermal-hydraulics in the core is modeled via a drift-flux model with flow reversal capability for nonequilibrium, nonhomogeneous flow through multiple (up to 200) parallel coolant channels.

Figure 8.1 Stability map for BWR4

The SBWR specific models implemented into RAMONA-4B are the isolation condenser (IC), standby liquid control system (SLCS), and local boron transport. The boron transport model solves the boron transport equation in each of thermal-hydraulic cells in the RPV by means of standard donor-cell differencing with flow reversal logics. The detailed description of the various RAMONA-4B models is given in Ref. [8.9] and that of its input descriptions is provided in Ref. [8.10]. Following important models required for a credible simulation of the SBWR startup transient are also available in RAMONA-4B.

- (1) Accurate low-pressure properties from 0.03 bar to 40 bar.
- (2) A heat slab model for stored energy in structure materials.
- (3) A flow-dependent loss coefficient model.
- (4) A Reactor Water Cleanup and Shutdown Cooling System (RWCU/SDC) model for level control during the SBWR startup.
- (5) A control rod drive (CRD) flow model.

- (6) A turbine bypass control model at pressures higher than 19 bar.

It should be mentioned that the heat transfer models for boiling and condensation at low pressures are not well established at present in the literature. Furthermore, the accuracy of these models at low pressures has not been validated for RAMONA-4B. Since the accuracy of these models may have a direct impact on the predicted phase change, thermal and flow characteristics, the results presented here should be considered qualitatively only.

8.2.2.2 Calculation Model for the Startup Transient

Simulation of startup conditions for the SBWR has been performed at BNL using the RAMONA-4B code. Three input models were developed, one each for SBWR-1200, SBWR-600 and SBWR-200 designs. Since information on the spatial and temporal movements of the control rods are not available from the GE report, in order to simulate the SBWR startup, a time-dependent power ramp profile as proposed by GE has been imposed as a boundary condition and a thermal-hydraulic only calculation performed with a simplified core model. The simplified core model is necessitated by the need to run the startup transient over a long period of time (3~6 hours) in order to show the important characteristics of the early part of the startup transient. The simplified core model consists of four parallel coolant channels including the bypass channel. The chimney, steam dome, downcomer, and the lower plenum were also included.

8.2.2.3 Initial Conditions for Startup

The initial conditions were selected to be as close to the actual values proposed by GE. Table 8.1 shows the initial conditions used in the present simulation.

Table 8.1 Initial Conditions for the Startup Transient

SBWR	Pressure (MPa)	Coolant Temp. (°C)	Core Power (MW)	Feedwater Flow (Kg/s)	Coolant Level (m)
SBWR-1200	0.055	40	0.04	0.0166	0.655*
SBWR-600	0.055	40	0.02	0.0083	0.655*
SBWR-200	0.055	40	0.00666	0.00276	0.655*

* Above the entrance to upper downcomer (18.17 m above core bottom)

8.2.2.4 Boundary Conditions for Startup

Table 8.2 presents the boundary conditions used for the startup simulation.

Table 8.2 Boundary Conditions for the Startup Simulation

Boundary Condition	Input to RAMONA-4B
Core Power	A ramp power of 20 MW/hr was imposed as a boundary condition, which corresponds to 42°C/s for SBWR-600 design. Similarly, 6.66 MW/hr was used for SBWR-200 design. For SBWR 1200, we used 40 MW/hr
Steam Flow	Turbine Control Valve (TCV) remained closed during the transient allowing reactor pressure to rise.
Feed water Temperature	Feed water heaters were inoperative since TCV was closed during the transient. Feed water temperature remained fixed at 80°C.

8.3 Additional Analyses

Model for 600MW for full core based on GE supplied information was reviewed. The model was tested for MSIV closure ATWS. In future start up procedure with pulling the control rod should be modeled. The procedure has to be developed.

Input for PUMA facility with RAMONA code was also prepared. USNRC has conducted startup tests with this facility to simulate possible startup for SBWR. A procedure was provided to run these start-up tests. The data is available to assess RAMONA code with this data. The input deck has been prepared and being debugged. The results of the analyses will be compared with the USNRC data.

8.4 Summary and Conclusions

Startup of the SBWR ranges from low pressure of 0.055 MPa to full pressure of 7.2 MPa. The most important modeling requirements for the startup transient are the low-pressure properties, the heat slab model for the stored energy, and the level control via the RWCU/SDC systems. Accurate modeling is also required for condensation and vapor generation at low pressures. Unfortunately, these models for low pressures are very scarce at best in the open literature and therefore, it is difficult to quantify the uncertainty in the predictions. However, we believe that the trend of the predictions by RAMONA-4B is qualitatively correct for the SBWR startup transient.

Three SBWR designs (SBWR-1200, SBWR-600 and SBWR-200) have been simulated for start-up transients and the results are similar. Neither the geysering instability nor the loop type instability

was predicted by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillation was not observed at all because the power level used in the simulation was not high enough. A study was made of the potential instability by imposing an unrealistically high power ramp in a short time period as suggested by GE. Core flow oscillations of small amplitude were predicted by RAMONA-4B with a period of between 31.8 s and 46.7 s similar to that of the TRACG prediction by GE.

Tables 8.3 and 8.4 show a comparison total core flow rate and vapor generation rates for three designs under normal and abnormal conditions. The flow parameters scale well with power as expected for the designs.

Table 8.3 Normal Operation

Power MW	Scale	Core Flow Rate Kg/s	Scale	Vapor Generation rate Kg/s	Scale
200	0.33	740	0.35	0.022	0.33
600	1	2125	1	0.066	1
1200	2.0	3800	1.8	0.14	2.1

Table 8.4 Abnormal Operation

Power MW	Scale	Core Flow Rate Kg/s	Scale	Vapor Generation rate Kg/s	Scale
200	0.33	1020	0.33	0.043	0.33
600	1	3100	1	0.13	1
1200	2.0	5900	1.9	0.26	2.0

The results of the RAMONA code predictions for SBWR-200 and SBWR-1200 for normal startup transient and abnormal startup transient are presented in figures 8.2-8.10.

8.5 References

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SBWR-1200MW NORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE MOD
 No RWCU Level Control (FRWC-1.2),
 EXIT MASS FLOW RATES AT CHANNEL 1, 2, 3, and 4

SBWR-200MW NORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE MODE
 No RWCU Level Control (FRWC-1.2), Pressure at 1.9 Bar
 EXIT MASS FLOW RATES AT CHANNEL 1, 2, 3, and 4

Figure 8.2 Channel flows

Figure 8.3 Total vapor generation rate in the vessel

SBWR-1200MW NORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE MOD
 No RWCU Level Control (FRWC=1.2),
 COLLAPSED WATER LEVEL RESPONSE

SBWR-200MW NORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE MODE
 No RWCU Level Control (FRWC=1.2), Pressure at 1.9 Bar
 MIXTURE LEVEL AND COLLAPSED WATER LEVEL RESPONSE

Figure 8.4 Liquid level above the downcomer entrance

Figure 8.5 Core power for abnormal start-up

Figure 8.6 System pressure for abnormal start-up

Figure 8.7 Core flow rate for abnormal start-up

SBWR-1200MW ABNORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE M
 No RWC Level Control (FRWC=1.2), Pressure at 1.9 Bar
 INLET MASS FLOW RATE AT CHANNEL 4

SBWR-200MW ABNORMAL STARTUP TRANSIENT 4-CHANNEL FULL CORE MO
 No RWC Level Control (FRWC=1.2), Pressure at 1.9 Bar
 INLET MASS FLOW RATE AT CHANNEL 4

Figure 8.8 Channel 4 flow rate for abnormal startup

Figure 8.9 Channel 1 flow rate for abnormal start-up

Figure 8.10 Total vapor generation rate for abnormal start-up

9. ACCOMPLISHMENTS

9.1 First Year Accomplishments

Here, the accomplishments of the first year are summarized.

- A preliminary design of the SBWR-200 and SBWR-1200 thermal-hydraulic systems were developed. The preliminary design involved identification of principle design criteria dictated by the safe operation of the reactor, identification of coolant requirements, design of the engineered safety systems, and emergency cooling systems based on passive systems.
- A large portion of the first year effort was concentrated on the design study of the SBWR-200
- Preliminary reactor engineered safety systems including reactor-cooling system and emergency core cooling systems were designed for SBWR-200 and SBWR-1200. These systems were based on a gravitational driving head. The passive safety system is reliable and safe to operate.
- A detailed scaling analysis was performed. The results of the scaling study were used in the performance and evaluation of the integral tests and data analysis. The scaling analysis identified key thermal-hydraulic parameters that govern flow phenomena in the SBWR. The analysis was based on the three-level scaling approach: (1) the integral response function scaling, (2) control volume and boundary flow scaling, and (3) local phenomena scaling. The first two levels are termed the *top-down* approach while the third level is the *bottom-up* approach. This scheme provides a scaling methodology that is practical and yields technically justifiable results. It ensures that both the steady state and dynamic conditions are simulated within each component, as well as the inter-component mass and energy flows, and the mass and energy inventories.
- Large break and small break LOCA integral tests for the SBWR-200 were carried out. These integral tests were performed to assess the safety systems and the response of the emergency core cooling systems. These tests were conducted in a scaled facility called PUMA located at Purdue University. The results have been used in the analysis of the safety systems. A main steam line break accident test was carried out for the large break loss of coolant DBA analysis while the bottom drain line break accident test was carried out for a small break loss of coolant DBA.

- RELAP5/MOD3 best estimate reactor thermal-hydraulic code was used to model the PUMA MSLB and BDLB integral tests. The analysis was used to demonstrate the safety features of the modular SBWR design and to validate the code applicability at the facility level. Overall, the code gave a reasonably accurate prediction of the system thermal-hydraulic behavior. This allows for an accurate assessment of the design of the SBWR-200 safety components. It also indicates a code deficiency that should be improved for a better simulation.
- A preliminary neutronics analysis and core design for SBWR-200 and SBWR-1200 have been performed. The neutronics work performed during the first year of the project has been: (1) to acquire and validate the computer codes required for the neutronics design and analysis of the SBWR (HELIOS, PARCS, and RELAP5/TRAC); (2) to develop neutronics and thermal-hydraulics models of the SBWR-600 and compare results to the RAMONA-4B predictions; and, (3) to perform preliminary designs of the SBWR-200 and SBWR-1200.
- A preliminary study on the stability of the SBWR-200 and SBWR-600 under normal startup and abnormal startup has been performed. Two SBWR designs (SBWR-600 and SBWR-200) have been simulated for start-up transients and the results were similar. Neither the geysering instability nor the loop type instability was predicted by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillations were not observed at all because the power level used in the simulation was not high enough to trigger these phenomena. An additional study was performed to investigate the potential instability by imposing an unrealistically high power ramp. RAMONA-4B predicted core flow oscillations of small amplitude, with a period between 31.8 s and 46.7 s. These predictions were similar to that of the TRACG prediction by GE.

9.2 Second Year Accomplishments

The accomplishments of the second year are summarized as follows:

- A design of the compact SBWR-200 and SBWR-1200 thermal-hydraulic system and neutronic systems were developed. The design involved identification of the principal design criteria dictated by the safe operation of the reactor, identification of coolant requirements, design of the engineered safety systems, emergency-cooling systems based on passive systems and scaling analyses.

- A large portion of the second year effort was concentrated on the design study of the SBWR-1200.
- A novel passive design of the hydraulic vacuum breaker check valve was developed and evaluated through RELAP5 simulation. This new check valve is based on the hydrostatic head, and has no moving components. It is comprised of only one additional tank, and one set of piping to the SP and DW. This system is simple in design and hence, easy to maintain and qualify for operation. The RELAP5 simulations were performed for the SBWR-200, with the old mechanical and new HVBC valve, for the MSLB transient. The new valve performance agreed quite well with that of the mechanical vacuum breaker check valve.
- The hydraulic vacuum breaker check valve was designed, constructed and installed for testing in the PUMA facility. A test was carried out on the PUMA with the new HVBC valve for the MSLB accident. The test data for the HVBC valve was compared with the data from MVBC valve, and the agreement was very good.
- A safety analysis for an anticipated transient with scram was performed for the SBWR-200 using RELAP5. The transient considered was a main steam isolation valve closure accident with scram. The simulation showed that the reactor is shut down without emergency water injection, and the decay heat is adequately removed by the ICS.
- Design basis accident scenarios were studied for the safety assessment of the SBWR-1200. Large break and small break LOCA integral tests for the SBWR-1200 were carried out. These integral tests were performed to assess the safety systems and the response of the emergency core cooling systems to a LOCA. They were conducted in a scaled facility called PUMA. The results have been used in the analysis of the safety systems. A main steam line break accident test was carried out for the large break loss of coolant DBA analysis while the bottom drain line break accident test was carried out for small break loss of coolant DBA.
- RELAP5/MOD3 best estimate reactor thermal-hydraulic code was used to model the PUMA MSLB and BDLB integral tests. The analysis was used to demonstrate the safety features of the modular SBWR design and to validate the code applicability in the facility scope. Overall, the code gave a reasonably accurate prediction of the system thermal-hydraulic behaviors. This allows for an accurate assessment of the design feature of the SBWR-1200

safety components. It also indicates some code deficiency that should be improved for a better simulation.

- Core depletion calculations were performed with PARCS, for a full fuel cycle analysis. A fuel lattice design was developed to optimize the fuel cycle safety parameters. A detailed study was carried out to improve the neutronics/thermal-hydraulics of all of the SBWR models. A preliminary fuel cycle analysis of the SBWR-200 and SBWR-1200 was carried out.
- A stability study of the SBWR-1200 under normal startup and abnormal startup has been performed. Neither the geysering instability nor the loop type instability was predicted by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillation was not observed at all because the power level used in the simulation was not high enough. A study was made of the potential instability by imposing an unrealistically high power ramp, with pressure restricted to 1.9 bar, as suggested by GE. Core flow oscillations of small amplitude were predicted by RAMONA-4B with a period of between 31.8 s and 46.7 s, similar to that of the TRACG prediction by GE.

9.3 Third Year Accomplishments

The accomplishments of the third year are summarized as follows:

- A design of the compact SBWR-200 and SBWR-1200 thermal-hydraulic system and neutronic systems were developed. The design involved identification of the principal design criteria dictated by the safe operation of the reactor, identification of coolant requirements, design of the engineered safety systems, emergency-cooling systems based on passive systems and scaling analyses.
- A large portion of the second year effort was concentrated on the design study of the SBWR-1200.
- A novel passive design of the hydraulic vacuum breaker check valve was developed and evaluated through RELAP5 simulation. This new check valve is based on the hydrostatic head, and has no moving components. It is comprised of only one additional tank, and one set of piping to the SP and DW. This system is simple in design and hence, easy to maintain and qualify for operation. The RELAP5 simulations were performed for the SBWR-200, with

the old mechanical and new HVBC valve, for the MSLB transient. The new valve performance agreed quite well with that of the mechanical vacuum breaker check valve.

- The hydraulic vacuum breaker check valve was designed, constructed and installed for testing in the PUMA facility. A test was carried out on the PUMA with the new HVBC valve for the MSLB accident. The test data for the HVBC valve was compared with the data from MVBC valve, and the agreement was very good.
- A safety analysis for an anticipated transient with scram was performed for the SBWR-200 using RELAP5. The transient considered was a main steam isolation valve closure accident with scram. The simulation showed that the reactor is shut down without emergency water injection, and the decay heat is adequately removed by the ICS.
- Design basis accident scenarios were studied for the safety assessment of the SBWR-1200. Large break and small break LOCA integral tests for the SBWR-1200 were carried out. These integral tests were performed to assess the safety systems and the response of the emergency core cooling systems to a LOCA. They were conducted in a scaled facility called PUMA. The results have been used in the analysis of the safety systems. A main steam line break accident test was carried out for the large break loss of coolant DBA analysis while the bottom drain line break accident test was carried out for small break loss of coolant DBA.
- RELAP5/MOD3 best estimate reactor thermal-hydraulic code was used to model the PUMA MSLB and BDLB integral tests. The analysis was used to demonstrate the safety features of the modular SBWR design and to validate the code applicability in the facility scope. Overall, the code gave a reasonably accurate prediction of the system thermal-hydraulic behaviors. This allows for an accurate assessment of the design feature of the SBWR-1200 safety components. It also indicates some code deficiency that should be improved for a better simulation.
- Core depletion calculations were performed with PARCS, for a full fuel cycle analysis. A fuel lattice design was developed to optimize the fuel cycle safety parameters. A detailed study was carried out to improve the neutronics/thermal-hydraulics of all of the SBWR models. A preliminary fuel cycle analysis of the SBWR-200 and SBWR-1200 was carried out.

- A stability study of the SBWR-1200 under normal startup and abnormal startup has been performed. Neither the geysering instability nor the loop type instability was predicted by RAMONA-4B in the startup simulation following the recommended procedure by GE. The density wave oscillation was not observed at all because the power level used in the simulation was not high enough. A study was made of the potential instability by imposing an unrealistically high power ramp, with pressure restricted to 1.9 bar, as suggested by GE. Core flow oscillations of small amplitude were predicted by RAMONA-4B with a period of between 31.8 s and 46.7 s, similar to that of the TRACG prediction by GE.