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Validation of Athlet with Purdue University Multi-Dimensional Integral Test Assembly (PUMA) Test

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VALIDATION OF ATHLET WITH

PURDUE UNIVERSITY MULTI-DIMENSIONAL

INTEGRAL TEST ASSEMBLY (PUMA) TEST

by

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ABSTRACT

Following the Three Mile Island Unit 2 (TMI-2) disaster, the development of a diagnostic system for emergency situations in a nuclear power plant was identified as an essential research area to strengthen the safety of nuclear power plants. A real-time estimation system of vital safety parameters on the main side of a PWR (Pressurized Water Reactor) facility was investigated since precise post-event scenario estimations are thought to be a crucial component of the diagnostic system. The alarm system of the plant can detect an SB-LOCA (Small Break Loss-Of-Coolant-Accident), but there were a couple of potential long-term transient occurrences. The accident situation might get worse if the facility is not provided with the appropriate, timely protection it required. It is necessary to correctly predict the size and position of a tiny break as well as important state variables inside the main side, such as the pressure, temperature, and flow rates of the various primary loops as well as the temperature distribution of the fuel rods. With the information provided, it was possible to choose the best course of action for protection. The objective of this document was to model the PUMA test facility in ATHLET to assess the thermal-hydraulic response of ATHLET. This article demonstrated and assessed ATHLET's capability to accurately forecast the minor break LOCA phenomenon after severe incidents.

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NOMENCLATURE

Acronyms	*Description*
RCS	Reactor Cooling System
RPV	Reactor Pressure Vessel
LWR	Light Water Reactor
GDCS	Gravity Drain Core Cooling System
KIT	Karlsruhe Institute of Technology
PCCS	Passive Containment Cooling System
ICS	Isolation Condenser System
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
LOCA	Loss of Coolant Accident
BWR	Boiling Water Reactor
PWR	Pressurized Water Reactor
SBWR	Simplified Boiling Water Reactor
VVER	Water-Water Energy Reactor
RELAP	Reactor Excursion and Leak Analysis Program
NRC	Nuclear Regulatory Commission
LPIS	Low-Pressure Injection System
ADS	Automatic Depressurization System
TRACE	TRAC/RELAP Advanced Computational Engine

Latin Characters	*Description*
α	steam void fraction (-)
w	velocity (m/s)
p	density (kg/m^3)
Ψ	interphase mass transfer rate (kg/s)
τ_i	interfacial shear per unit volume (N/m3)
H	elevation [m]
g	gravity constant (m/s^2)
S	momentum source (N/m^3)
f_{wall}	wall friction force per unit volume (N/m^3)
D	hydraulic diameter (m)
A	(flow) area [m^2]
j	superficial velocity (m/s)
q	heat flux (W/m^2)
C_o	phase distribution parameter
W	specific heat generation rate (W/m^3)
c_p	specific heat capacity at constant pressure (J/kg/K)
T	Temperature (^oC, K)
λ	Darcy-Weisbach friction factor (-) heat conductivity (W/m/K)
V	volume (m^3)
t	time (s)

Subscripts, superscripts	*Description*
V,v	vapor (or gas, resp.)
L,l	liquid
i	interphase
R,r	relative
W	wall
m	mixture
Γ	interphase mass exchang

1. INTRODUCTION

Following the Three Mile Island Unit 2 (TMI-2) disaster, the development of a diagnostic system for emergency situations in a nuclear power plant was identified as an essential research area to strengthen the safety of nuclear power plants (Prošek & Mavko, 1999). A real-time estimation system of vital safety parameters on the main side of a PWR (Pressurized Water Reactor) facility was investigated because precise post-event scenario estimations are thought to be a crucial component of the diagnostic system. The alarm system of the plant can detect an Small Break Loss of Coolant Accident (SB-LOCA), but there are a few long-term transitory occurrences that could occur (Karwat, 1985). The accident situation might easily get worse if the facility does not provide the appropriate, timely protection it requires. It is necessary to correctly predict the size, position of a tiny break, and important state variables inside the main side, such as pressure, temperature, and flow rates of the various primary loops as well as the temperature distribution of the fuel rods. It would be possible to successfully decide on the best course of action for protection using this knowledge. The system pressure serves as a proxy for identifying a transient period of SB-LOCA. Normally, the Reactor Cooling System (RCS) pressure will rapidly drop when the break begins to occur during a blowdown. When the pressure is stabilized for a prolonged period during natural circulation, blowdown has ended. The boiloff period, which is characterized by a monotonically falling system pressure, starts when the loop seal is cleared. While core recovery occurs, the pressure is stabilized to a new quasi-equilibrium that is dictated by the flow resistances in the system and at the break (Frepoli, 2006).

Reactor Excursion and Leak Analysis Program (RELAP) is a simulation program that enables users to simulate different operating transients and speculated accidents that could happen in a nuclear reactor. It is accomplished by coordinating the behavior of the core with that of the cooling system of the reactor. Idaho National Laboratory created RELAP to meet the urgent requirement for reactor safety studies, and it is still in use today. Universities can utilize RELAP as a teaching tool or for reactor safety studies, reactor design, operator simulator training, and more (Prošek & Mavko, 1999). To assess transient and steady-state thermal-hydraulic behavior in Light Water Reactors (LWRs), the US Nuclear Regulatory Commission (NRC) created the reactor systems code TRAC/RELAP Advanced Computational Engine (TRACE). Additionally, best-estimate studies of Loss of Coolant Accident (LOCA), operational transients, and other accident scenarios in PWRs and Boiling Water Reactors (BWRs) may be performed using this program (Kawanishi et al., 1991). ATHLET (Analysis of Thermal-hydraulics of Leaks and Transients) is a sophisticated best-estimate system coding initially developed to perform modeling of engineering basis and beyond-design basis catastrophes (all without core deterioration) in light-water nuclear reactors, including RBMK and Water-Water Energy Reactor (VVER) (U. Gaal et al., 1986). The version of this program also permits the modeling of operations with fluids such as helium, liquid metals, and molten compounds.

As of 2023, the international community is beginning to develop novel, cutting-edge reactor designs while considering the suitability of nuclear technology in emerging and industrialized countries' energy mix plans (Lee & Ishii, 1990). There are a couple of facilities to develop analysis in nuclear power plants like PUMA (Lee & Ishii, 1990), PANDA (Huggenberger, et al. 2000), APEX (Lee & Ishii, 1990), and MASLWR (Han, et al. 1993). For research on the security of modern and advanced LWRs, the PANDA facility is a sizable, multicompartmental thermal hydraulic facility. The facility is multifunctional, and the applications include distinct effect tests, component tests, fundamental system tests, and integral containment reaction tests (Ishii, M., et al., 2006). The APEX test facility has a one-fourth height and one-half time scale, integrated systems, and decreased pressure (Pimentel, 1996). A thorough scaling study was carried out to guarantee that it accurately replicated the details of the AP600 geometry, including the primary system, the passive safety systems, and elements of the non-safety grade Chemical and Volume Control System (CVS) and Residual Heat Removal System (Pimentel, 1996). Oregon State University (OSU) has constructed a small modular Pressurized Water Reactor (PWR) with the Multi-application Small Light Water Reactor (MASLWR) design that depends on natural circulation during both steady-state and transient operation (Pimentel, 1996). Using the PUMA experiment, the overall efficacy of the GDCS (Gravity Drain Core Cooling System), PCCS (Passive Containment Cooling System), and Simple Boiling Water Reactor (SBWR) phenomena critical to LOCAs and other transients were evaluated. The design of the scaled facility was created using a three-level scaling procedure. General Electric (GE) created the SBWR, a brand-new boiling water reactor (Ishii, M., et al., 2006). Current Boiling Water Reactors and SBWRs are distinguished primarily by the incorporation of passive emergency cooling devices and the simplification of the coolant circulation system (Leonardi, et al. 1998). The SBWR's vessel does not have recirculation pumps to circulate the coolant. There are no active pump-injected flows in the containment cooling system or emergency core cooling system (Ishii, M., et al., 2006).

1.1 RESEARCH OBJECTIVES

The purpose of this study was to better understand the features of a small break loss-of-coolant accident analysis for the safety injection system of the PUMA (Purdue University Multi-Dimensional Integral Test Assembly).

- Simulate the test of PUMA (Purdue University Multi-Dimensional Integral Test Assembly) in the ATHLET simulation code to evaluate the thermal-hydraulic results of the ATHLET.

- Evaluate the materiality of the ATHLET in estimating small break LOCA phenomena during LOCA accidents.

1.2 LITERATURE REVIEW

This section includes a literature review of Small Break LOCA and Purdue University Multi-Dimensional Integral Test Assembly (PUMA).

1.2.1 Small Break LOCA. A Small Break Loss-of-Coolant Accident (SBLOCA) is a critical event in the safety analysis of nuclear power plants. It involves a breach in a small pipe or component within the primary coolant circuit, leading to the gradual loss of reactor coolant (Prošek & Mavko, 1999). Research in this area has focused on understanding the behavior of SBLOCA, its impact on reactor safety, and the effectiveness of safety systems and strategies to mitigate its consequences. Studies have investigated the thermal-hydraulic aspects of SBLOCA events, exploring phenomena like two-phase flow, heat transfer, and pressure drops (Prošek & Mavko, 1999). The progression of multi-phase flow instabilities, such as flow oscillations and chugging, has been examined to predict the behavior of the reactor coolant system during such events. Passive safety systems, which operate without external power, have gained attention for their reliability in maintaining cooling even if active systems fail. Researchers have assessed the performance of these systems in various SBLOCA scenarios to ensure the core remains adequately cooled (Prošek & Mavko, 1999).

1.2.1 PUMA. The Purdue University Multi-Dimensional Integral Test Assembly (PUMA) is a significant experimental facility that plays a pivotal role in advancing our understanding of nuclear reactor thermal hydraulics (Ishii, M., et al., 2006). Through its contributions to code validation, two-phase flow studies, accident simulations, and more, PUMA continues to be a crucial resource for ensuring the safety and efficiency of nuclear power plants (Ishii, M., et al., 2006).

2. BACKGROUND

2.1 SMALL BREAK LOCA

The LOCA (Loss-Of-Coolant Accident) of a nuclear reactor is a type of failure that, if it is not handled carefully, might lead to reactor core damage. The Emergency Core Cooling System (ECCS) of each nuclear facility is designed to respond to a LOCA.

Small breaks are defined as those with flow regions typically more than 3/8 in. in diameter and area less than 1 ft^2 (Karwat, 1985). The High-Pressure Safety Injection (HPSI) system is automatically starts when a tiny rupture is sufficiently large to cause the primary system to depressurize to the high-pressure safety injection set point. Because the reactor charge flow can replenish the lost inventory, breaks smaller than 3/8-inch in diameter do not cause the reactor coolant system to lose pressure (Prošek & Mavko, 1999). Only decay heat is produced in the core of the reactor once the control rods shut down the machine. The interaction among the core power level, the axial power shape, the break size, the high-head safety injection performance, and the pressure at which the accumulator begins to inject determines the limiting small-break LOCA. [9The limiting break, which must be both big and small enough to prevent the reactor system from swiftly depressurizing to the accumulator set point, stops the high-pressure safety injection system from making up for the mass of the reactor system. A core recovery is the outcome of this convergence of circumstances (Boyack, B.E., et al. 2001).

For the purposes of this study, the SBLOCA (Small Break LOCA) was modeled in the ATHLET. This LOCA transient was initiated by an immediate break in one of the GDCS injection lines. As a consequence of a double-ended pipe break, the ruptured flow is released through the drywell. Flow limitation apertures attached to every GDCS injector line at the RPV (Reactor Pressure Vessel) outlet capture the break flow exiting the reactor vessel (Lim, et al. 2014). The lower RPV elevation is linked to the GDCS discharge conduit. A substantial amount of GDCS and RPV fluids may escape the containment, bringing the RPV water level close to the TAF (Karwat, 1985).

To study fundamental phenomenology during SB-LOCA, the scenarios were simple. While the logic of the protection system was modeled, no operator actions were specified in the scenarios, except for the reactor coolant pump trip in compliance with emergency operating regulations. When the protection system's logic detects a hazard, the safety system must be activated. There are two separate, redundant protection trains. Only the safety systems attached to a protection train are activated. To decrease the impact of probable design-based accidents, safety measures are implemented (Prošek & Mavko, 1999). The most important safety component for SB LOCA is the emergency core cooling system, which is designed to cool the core (Parzer, 2001). The components of this system are an accumulator and a low-pressure safety injection (LPSI) pump, and a high-pressure safety injection (HPSI) pump. The pumps transfer water from a tank used to store refueling water to the reactor vessel. AFW (Auxiliary Feedwater) is also a crucial safety system. It supplies water to the secondary part of the vapor engine so that the heat sink can be maintained.

The decay heat generated in the reactor core may be removed when a small break develops in the reactor coolant system by moving heat to the secondary side of steam generators, injecting emergency core cooling, and releasing heat with break flow (Jeong, 2002). When the break size is large, the break flow is largely responsible for removing heat. However, steam generator cooling is more important when the break size decreases. Following the commencement of an SB-LOCA, single-phase or two-phase natural circulation is used to transport heat through steam generators when the RCS is properly supplied with water. The natural circulation is stopped as primary coolant levels drop, and steam starts to condense in steam generator tubes before returning via hot legs to the reactor vessel. The term "reflux condensation phenomena" is used to describe this phenomenon (Kawanishi et al., 1991). In the case of SB-LOCA, the system spends more time in the higher-pressure area as the reactor vessel gradually loses pressure. Building and maintaining a high-pressure system is substantially more expensive than one that operates at low pressure. Finding a scaling rationale that will enable a low-pressure test facility to mimic an SB-LOCA in the high-pressure prototype is therefore motivated (Karwat, 1985).

ATHLET simulations of SB-LOCA led to the development of models for analyzing various SB-LOCA transient scenarios. These scenarios included additional triggering logic for various systems for the situation with presumed loss of off-site power as well as break models for the two chosen breaks. The simulation was stopped after about 60000 seconds for a scenario with a 2-break size of 4 and 6 cm because of the oscillating behavior of some basic system parameters when the LPIS (Low-Pressure Injection System) was operating. As a result, a few minor adjustments were made to the original input model. A reduction in a few junction areas in the reactor vessel was among these modifications, which was followed by the normal completion of the calculation.

2.2 PUMA

The purpose of the PUMA experiment was to evaluate the integral effectiveness of the GDCS and PCCS, as well as the SBWR phenomena pertinent to LOCAs and other interruptions (Leonardi, et al. 1998). The scaled facility design was made using a three-level scaling process. The foundation of the first level is the integral scaling technique, which was developed from the integral response function (RAVANKAR, et al. 1996). This level guarantees that the steady-state and dynamic properties of the loops are scaled effectively. To guarantee that the flow and inventory are accurate, the boundary flow of mass and energy between components is scaled on a second level. The main local phenomena and constitutive links are the focus of the third level. The building is scaled at 1/4 height and 1/100 area ratio, translating to a volume size of 1/400 (RAVANKAR, et al. 1996). The power scaling, according to integral scaling, is 1/200 (RAVANKAR, et al. 1996). The present scaling method forecasts that time will pass through the model twice as rapidly as first level. The following scram, PUMA was designed to run at and below 150 Psia and is scaled for maximum pressure (RAVANKAR, et al. 1996). All of the major SBWR (Simplified Boiling Water Reactor) safety and non-safety systems that are substantial to transients are modeled at the site. This study presented the complete instrumentations and model component designs.

After determining the assessment facility scalability based on integral and boundary flow expansion, the third stage of scalability for localized processes is evaluated. The first two degrees of scaling can be met while still experiencing certain aberrations in local phenomena (Ishii, M., et al., 2006). Particular phenomena should be scaled as precisely as possible within the constraints of the two prior scaling levels, as the first two scaling stages must be achieved for a facility to be entirely scaled. Regional problems relevant to SBWR consist of choking (blowdown), flashing, forced or natural circulation patterns, bypass flow in the reactor core, critical heat flux, slip and phase distribution (flow regime), thermal stratification in the suppression pool, condensation, mixing, wall stored energy, and heat loss (Kocamustafaogullari & Ishii, 1983).

For an efficient layout of the scaled-down SBWR integral test establishment, PUMA, it is necessary to develop an equal and reasonable scaling strategy. The bottom-up approach focuses on the scaling of significant local phenomena, whereas the top-down approach concentrates on the scaling of integral systems. Together, these two approaches form the foundation of PUMA scaling. The NRC Technical Program Group devised a theoretical structure for this scalability technique for serious disaster scalability. Together these strategies provide a useful scalability technique that yields scientifically supported results. The three-level scaling methodology is what makes up the actual PUMA scaling method. The integral system scaling (which is also referred to as universal scaling or the top-down method) consists of two phases: the integral reaction function scaling on the first level, and the regulation capacity and border flow scalability on the second stage. The third level of scaling, local phenomenon scaling, implements the bottom-up methodology. The transient response functions for the key variables in single- and two-phase flow are used to derive the substantial portion of the integral system scaling. This scaling ensures that each component accurately simulates the two steady-state and dynamic conditions. The significant thermal-hydraulic parameters are simulated as a result of the integral response function scaling. The relationships between the various components must be scaled appropriately for a system like the SBWR, which has numerous components. The scaling strategy should be founded on principles of conservation. Utilizing the energy and mass control capacity balance formulas, the important scaling parameters for the inter-component relations are determined. These requirements measure both the amount of mass and energy inventories within each component and the mass and energy fluxes between components; thus, they are essential to the scaling of the entire system (GE Nuclear Energy, 1992).

After structure scalability is complete, the academic concept for the system must be executed. Many practical factors, including the instruments, must be considered at this stage. The research concept must then be transformed into a design for engineering that adheres to specifications such as local and state licensing laws, manufacturability requirements, test facility operation and maintenance requirements, and etc. Appropriately sized integral testing equipment will generate pertinent integral scientific information that replicates the most significant phenomena of relevance. However, neither the engineering nor the academic designs are capable of satisfying all scaling conditions (Lim, et al. 2014). Consequently, scale anomalies are inevitable, particularly at the third stage of scaling. Two variables account for most distortions: trouble in meeting the regional scaling requirements and ignorance of the pertinent localized phenomena. Thus, straightforward extrapolation of the results of experiments to prototypical conditions is typically extremely difficult or even impossible (Han, et al. 1993).

Since the PANDA and PUMA systems were converted to scale by the ESBWR using the same scaling method but varying scaling percentages, it is possible to conduct equivalent testing between them. Since both facilities utilize full-pressure scaling, the same thermodynamic fluid characteristics will be maintained. The PANDA facility, unlike the PUMA facility, uses an exhaustive height scaling (Huggenberger, et al. 2000).

2.2.1 Design Basis for the Puma Integral Facility. PUMA's scaling strategy is based on three stages of scaling: integral structure scaling, energy and mass accounting and boundary flow scaling, and scaling of local phenomena. In the integral system scaling, the fluid consistency, continuous momentum, and energy formulas are employed alongside the right boundary requirements and the solid equation for energy. The dimensionless version of integral response functions can be used to construct without dimensions groups that characterize geometrical, kinematic, dynamic, and energy similarities. The friction number, Richardson number, characteristic time constant ratio, Biot, and heat source number are among the geometrical non-dimensional groups. For a two-phase system, the dimensionless numbers are the Zuber, subcooling, Froude, drift-flux, thermal inertia ratio, two-phase friction, and orifice (Ishii, M., et al., 2006).

The total energy and mass regulation capacity balance formulae are applied to boundary flow scaling and mass and energy inventories. Using the proportional standards, the flow region of the channel, velocity of the fluid, and enthalpy flow are identical (Ishii, M., et al., 2006).

The prominent phenomena are investigated separately, and suitable similarity parameters are calculated at the scale of local phenomena. Flow fluctuations local phenomena considered include constricted flow, unchoked flow, flow regime, relative velocity, critical heat flux, spontaneous circulation, flashing, condensation, heat source, sink, mixing, and stratification (Ishii, M., et al., 2006).

The scaling parameters for the verified integral evaluation establishment were determined by balancing a number of criteria. For instance, it is necessary to build a facility that is both large enough to serve as a useful starting point for extrapolating the full-scale prototype and manageable in terms of size and expense. This compromise involved building a construction that was scaled down by 1/4 in height and 1/400 in volume (GE Nuclear Energy, 1992).

A list of the benefits of the current scale model is as follows.

1. A realistic simulation of both the hydrostatic driving head and frictional resistance. To maintain the loop's thermal-hydraulic properties, this balance is crucial (GE Nuclear Energy, 1992).

2. A small aspect ratio resembling the system's prototype. Just 2.5 is the value of /RAIR compared to a linear scale of 1. More than any other building that was built or is currently being built, this value of 2.5 is closest to unity. Thus, the two-dimensional and three-dimensional phenomena are depicted correctly (GE Nuclear Energy, 1992).

3. A minimal amount of heat is lost from the structure when the model of diminished height is used. For a fixed-volume size model, a shorter facility results in less heat loss distortion. The 1/4 height reduction provides a substantial advantage compared to a full-height structure (GE Nuclear Energy, 1992).

4. Modeling with the exactitude of hydrostatic head-driven flow rates in conduits. Using the supplied scaling method, the flow rates in PCCS, GDCS, and additional connecting lines are accurately adjusted to simulate the prototype conditions. According

to the scaling study, the entire height is not necessary. Contrary to popular assumption, the finest simulation does not require full-height simulation (GE Nuclear Energy, 1992).

5. As the liquid elevates in the RPV, flashing phenomena persist due to a decrease in the hydrostatic head. A comprehensive analysis of the flashing phenomenon indicates that a height reduction greater than 1/4 scale diminishing the liquid flashing phenomenon can start significantly. Given that mixture level is one of the most crucial safety considerations, a 1/4-inch height seems reasonable. (GE Nuclear Energy, 1992).

2.2.2 Puma System Design. In this part, the SBWR system parameters that are crucial to the reactor's safety are discussed. GE's Standard Safety Analysis Report (SSAR) provided some of the data tables used in the debate. All the dimensions given correspond to nominal size (Lee & Ishii, 1990).

The cross-sectional illustration of the SBWR vessel is displayed in Table 2.1.

The thermal-hydraulic parameters of the RPV during normal full-power operation are displayed in Table 2.2.

Figures 2.1,2.2 and 2.3 show the PUMA integral facilities used in different studies.

Table 2.3 shows the line parameters of the Gravity Driven Core Cooling System.

Table 2.1 The Size of Reactor Pressure Vessel (Lee & Ishii, 1990).

Inside Height	24.505 m
ID	6m
Wall Thickness	157.175 mm
Coolant Volume	607.3 m3
Total Volume	669 m3
Active Fuel Length	2.743 m

Table 2.2 Thermal-Hydraulic Parameters for RPV (Lee & Ishii, 1990).

Core Power (100%)	2000 MWth
Core Inlet Flow	27.2 x 106 kg/h
Feedwater Inlet Flow	3.88 x 106 kg/h
Steam Dome Pressure	7.17 MPa
Core Inlet Pressure	7.28 MPa
Core Outlet Pressure	7.23 MPa
Average Core Power Density	41.5 kW/liter
Average Heat Flux	430.58 kW/m2
Maximum Heat Flux	1225.23 kW/m2
Core Average Exit Quality	14.3
Feedwater Temperature	215.6°C
Core Inlet Temperature	278.5°C
Core Outlet Temperature	288°C

Table 2.3 Line Parameters for GDCS (Lee & Ishii, 1990).

GDCS Pool Numbers	3
Each GDCS Pool Minimum Drainable Inventory	329 m^3
Minimum Surface Elevation of GDCS	13.3 m
GDCS Injection Line Nozzle Size at RPV	146.3/6 mm/inch
Equalization Line Nozzle Size at RPV	76.2 mm
Water Level of GDCS	1321 mm

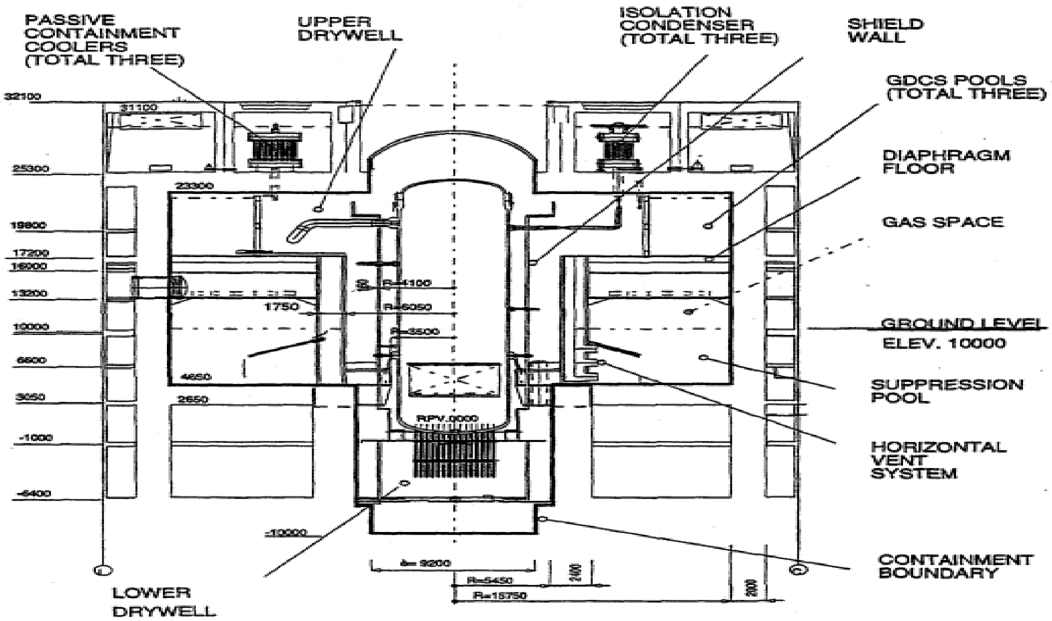


Figure 2.1 PUMA Integral Facility Containment (GE Nuclear Energy, 1992).

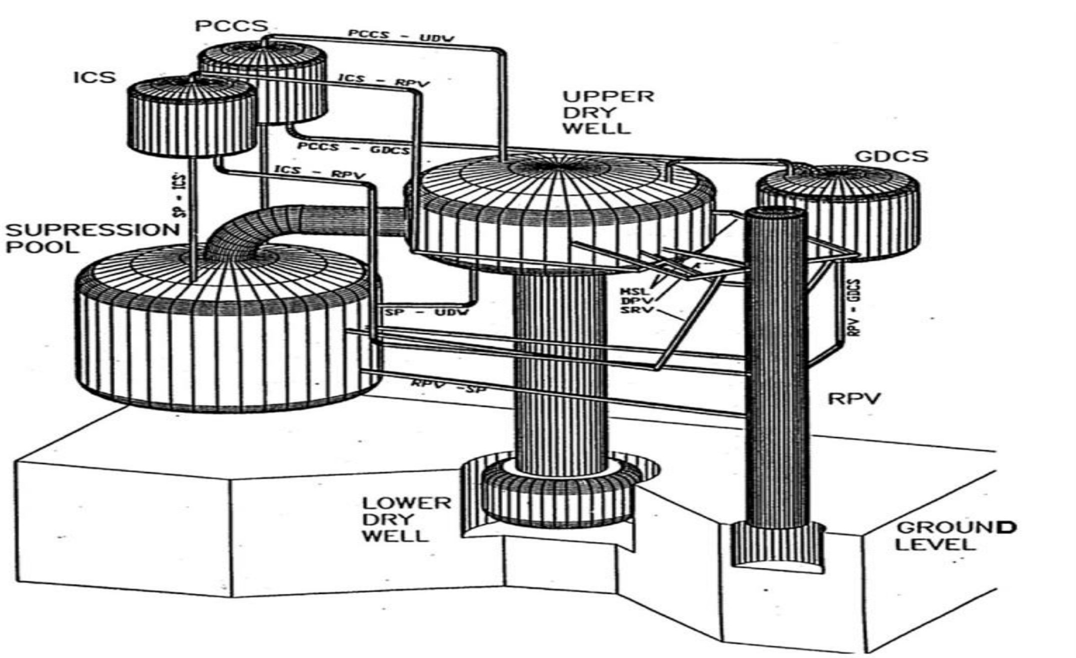


Figure 2.2 PUMA Integral Facility Containment (Lim, et al. 2014).

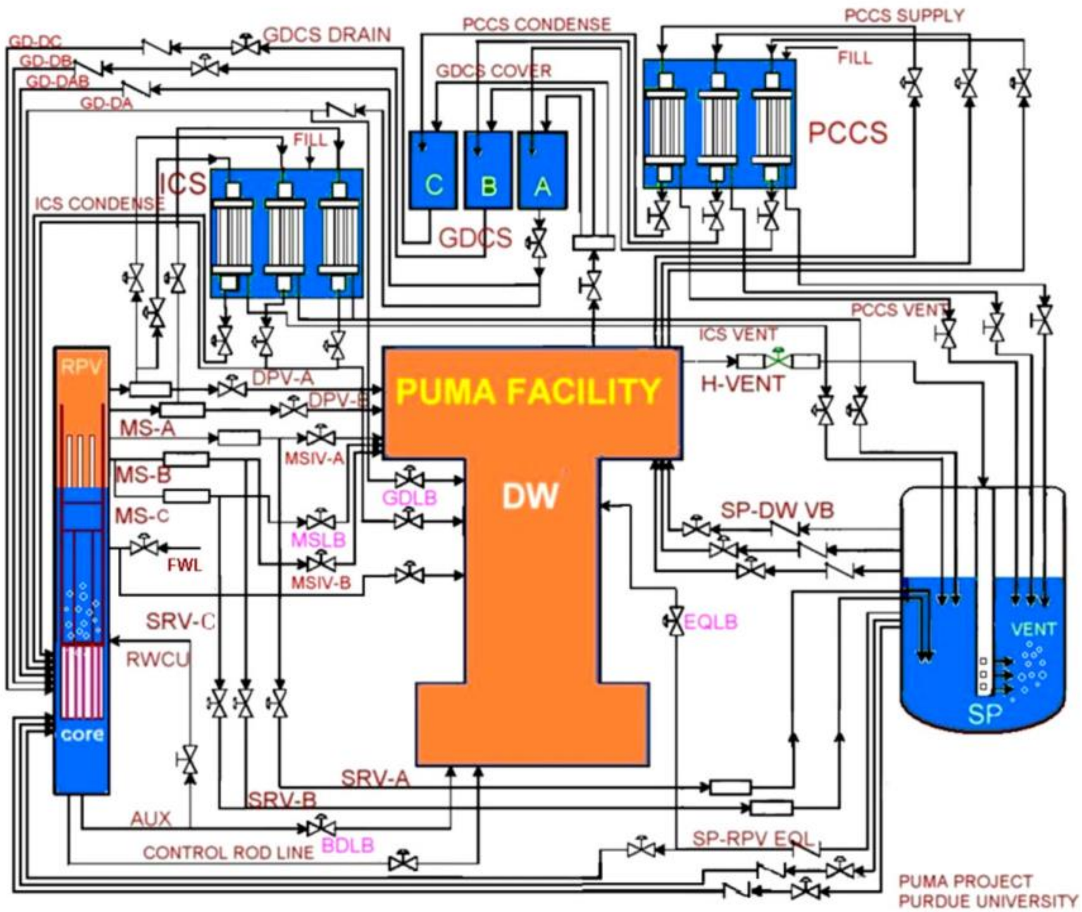


Figure 2.3 PUMA Integral Facility Containment (Ishii, et al. 2006).

2.3 ATHLET

ATHLET is a sophisticated best-estimate system code that was initially created for the modeling of design basis and beyond design basis accidents in light water reactors, including the VVER and RBMK reactors, without core deterioration. This program version also allows for the simulation of additional working fluids, such as helium, liquid metals, or molten salts (Vojacek & Mazzini, 2014).

The one-dimensional, two-phase fluid dynamic models are based on a five-equation model, while the full-range drift-flux model with a dynamic mixture-level tracking capability is an addition to this model (Vojacek & Mazzini, 2014). Also included is a two-fluid model built on six conservation equations. The heat conduction and heat transfer module make it possible to simulate fuel rods and structures in a variety of ways. One-dimensional kinetics models or point-kinetics are used to calculate nuclear heat generation. A comprehensive control simulation application is available for customizable emulation of BOP and supplemental plant structures (Vojacek & Mazzini, 2014).

3. METHOD

3.1 NODALIZATION

Figure 3.1 displays the nodalization scheme utilized by the PUMA facility within the ATHLET simulation code. The nodalization scheme of the ATHLET code includes components such as RPV, ICS (Isolation Condenser System), GDCS, PCCS, Upper and Lower Dry Well, and Sup-Pool.

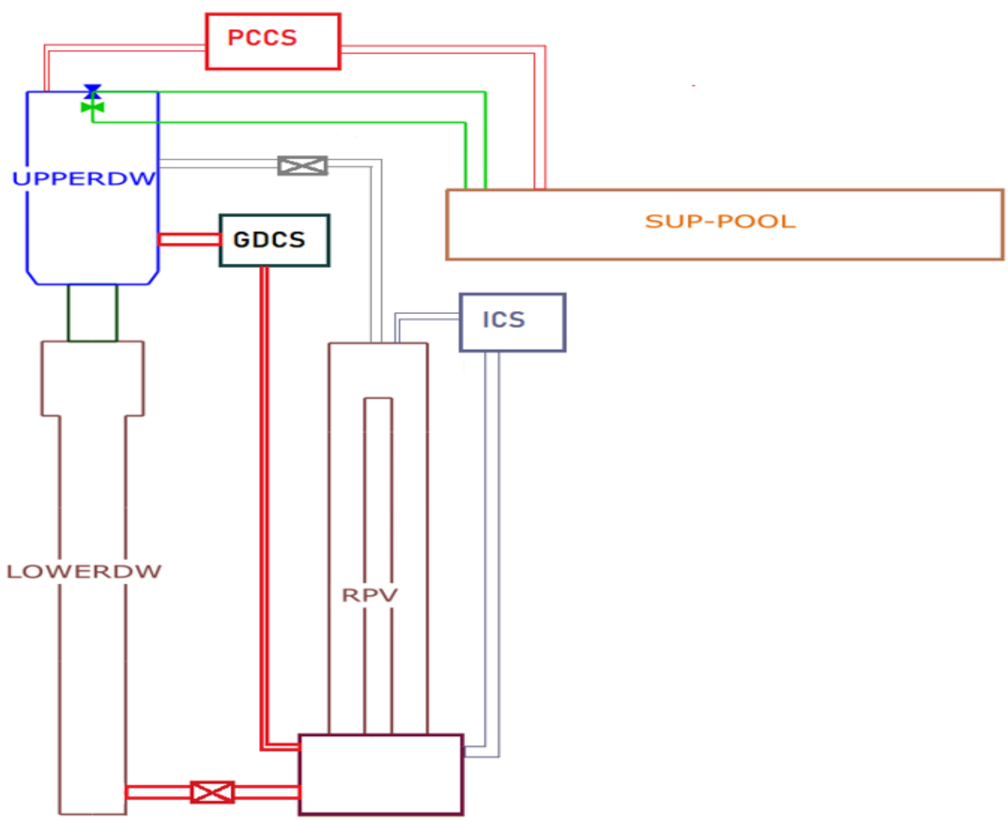


Figure 2.1 Nodalization Scheme.

3.2 MODULES

The Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) is in the process of developing the thermal-hydraulic computational code ATHLET for the evaluation of operational conditions, anomalous transients, and all varieties of leaks and breaches in nuclear power facilities (U. Gaal et al., 1986). The goal of the code development is to provide a single code that can handle all design basis and beyond-design basis incidents for PWRs, BWRs, SMRs, and future Gen IV reactors without core deterioration (Vojacek & Mazzini, 2014). The code is made up of a number of fundamental modules for the calculation of various phenomena, with an emphasis on how a nuclear power reactor operates (U. Gaal et al., 1986):

3.2.1 Thermo-Fluid dynamics (TFD). A modular network method is used by the ATHLET TFD module to model a thermal-hydraulic system. Simple thermo-fluid dynamic objects (TFOs), or basic fluid dynamic elements, can be connected to model any system configuration (Vojacek & Mazzini, 2014). Each of the many TFO kinds was used with a particular fluid dynamic model. All item kinds were divided into three fundamental groups: A modular network method is used by the ATHLET TFD module to model a thermal-hydraulic system. Simple thermo-fluid dynamic objects (TFOs), or basic fluid dynamic elements, can be connected to model any system configuration. Each of the many TFO kinds was used with a particular fluid dynamic model. All item kinds were divided into three fundamental groups:

- Pipe objects use a one-dimensional TFD model to describe the flow of fluid through. A pipe object can be thought of as a collection of sequential nodes (control volumes) connected by flow pathways (junctions) after being nodanized in accordance with the input data. A single junction pipe is a unique application of a pipe object that has just one junction and no control volumes (U. Gaal et al., 1986).

- Branch objects just one control volume is present. They use non-linear ordinary differential equations or algebraic equations in a zero-dimensional TFD model (U. Gaal et al., 1986).
- Special objects are used for network elements that have complicated geometries, such as the cross-connection of parallel pipe objects to create a multidimensional network (U. Gaal et al., 1986).
- Generally, the core degradation modules are coupled with the 5 equation TFD module that "ATHLET contains the conservation laws for vapor mass, liquid mass, vapor energy, liquid energy, and overall momentum."

Liquid Mass: $\frac{\partial((1-\alpha)p_L)}{\partial t} + \nabla \cdot \left((1-\alpha)\overrightarrow{w_L}pL\right) = -\Psi.$

Vapor Mass: $\frac{\partial(\alpha \cdot Pv)}{\partial t} + \nabla \cdot (\alpha \cdot \vec{w}_v pv) = \Psi.$

Liquid energy:

$$\frac{\partial\left[(1-\alpha)\rho_L\left(h_L+\frac{1}{2}\vec{w}_L\vec{w}_L-\frac{p}{\rho_L}\right)\right]}{\partial t} + \nabla \cdot \left[(1-\alpha)p_L\vec{w}_l\left(h_L+\frac{1}{2}\vec{w}_L\vec{w}_L\right)\right] = -\rho\frac{\partial(1-\alpha)}{\partial t}$$

Vapor energy:

$$\frac{\partial\left[\alpha\rho_v h_v+\frac{1}{2}\overrightarrow{w_v w_v}-\frac{\rho}{\rho_v}\right]}{\partial t} + \nabla \cdot \left[\alpha\rho_v\vec{w}_v\left(h_v+\frac{1}{2}\overrightarrow{w_v w_v}\right)\right] = -\rho\frac{\partial\alpha}{\partial t}.$$

Liquid momentum: $\frac{\partial[(1-\alpha)\rho_L\vec{w}_L]}{\partial t} + \nabla[(1-\alpha)\rho_L\vec{w}_L\vec{w}_L] + \nabla \cdot [(1-\alpha) \cdot p].$

Vapor momentum: $\frac{\partial(\alpha\rho_v\vec{w}_v)}{\partial t} + \nabla(\alpha\rho_v\vec{w}_v\vec{w}_v) + \nabla(\alpha \cdot p).$

The total momentum equation for the two-phase mixture is as follows:

$$\frac{\partial(\rho_m\vec{w}_n)}{\partial t} - \vec{w}_m\frac{\partial\rho_m}{\partial t} + \rho_m\vec{w}_m\nabla\vec{w}_m + \nabla\left[\alpha(1-\alpha)\frac{\rho_v\rho_L}{\rho_m}\vec{w}_R\vec{w}_R\right] + \nabla p.$$

3.2.2 Numerical Methods. The thermo-fluid dynamic model of temporal integration was carried out using the all-purpose ODE-solver FEBE (Forward-Euler, Backward-Euler). By breaking the linear system of first-order ordinary differential equations (ODE) into two subsystems, the first of which is explicitly integrated while the second of one is implicitly integrated to, yield the solution. In ATHLET, the totally implicit option is typically employed. The whole ODE system is provided by each thermo-fluid dynamic object, which is integrated by FEBE (U. Gaal et al., 1986).

3.2.3 Heat Conduction and Transfer. The fundamental module HECU simulates heat conduction in buildings, heat exchangers, fuel rods, electrical heaters, and spheres (pebble bed). Heat Conduction Objects (HCOs) may be assigned to each Thermal-Fluid dynamic entity within a specific network. The conservation of energy in a control volume was used to calculate the following heat conduction equation (Vojacek & Mazzini, 2014):

$$\int_V W\,dV = c_\rho \cdot \rho \cdot \int_V \frac{\delta T}{\delta t}dV + \int_S \vec{q}\,dA.$$

Rate of heat Rate of change of heat flow crossing

Generation internal energy the boundary

Then the heat flux can be described by the equation:

$$\int_S \vec{q}\, d\vec{A} = \lambda \cdot \int_S gradT \cdot d\vec{A}.$$

Observing the Gaussian rule, the right side of the equation can be transformed:

$$-\lambda \int_S gra\, dT \cdot d\vec{A} = -\lambda \int_V div(gra\, dT) \cdot dV = -\lambda \int_V \nabla^2 T \cdot dV.$$

Putting the above equation into the original one:

$$\int_V W \cdot dV = c_\rho \cdot \rho \cdot \int_V \frac{\delta T}{\delta t} dV - \lambda \int_V \nabla^2 T\, dV.$$

$$\frac{\delta T}{\delta t} = \frac{\lambda}{c_\rho \cdot \rho} \nabla^2 T + \frac{1}{c_\rho \cdot \rho} \cdot w.$$

This differential equation is the well-known Fourier equation (Vojacek & Mazzini, 2014).

3.2.4 Nuclear Heat Generation. The neutron kinetics module NEUKIN is typically used to model nuclear heat generation. For the simulation of electrically heated rods or for a straightforward, reduced description of a reactor core, it is optional to provide the entire amount of power produced as a function of time or any other variable. Nuclear reactors produce two distinct types of energy: thermal energy from the decomposition of long-lived fission products and prompt energy coming from fission and disintegration of short-lived products of fission. A GCSM signal is utilized to provide the constant portion of decay heat and its dependent on time decrease following a reactor scram. One-dimensional neutron dynamics or point-kinetics models can be used to determine the dependent on time behavior of rapid power production. It is expected that a certain percentage of the total power is created directly in the coolant rather than in the fuel (U. Gaal et al., 1986).

3.3 SCENARIOS

In this simulation experiment, a range of two different break sizes was accounted for with cracks in diameter and occurs in the reactor coolant system's cold leg. These breaches imply that water in the reactor vessel evaporates, and that reactor coolant is released through an aperture. This incident is referred to as a coolant loss accident. ATHLET simulations of SB-LOCA led to the development of models for analyzing various SB-LOCA transient scenarios. These scenarios included additional triggering logic for various systems for the situation with presumed loss of off-site power as well as break models for the two chosen breaks. The simulation was stopped after about 60000 seconds for a scenario with a 2-break size of 4 and 6 cm because of the oscillating behavior of some basic system parameters when the LPIS was operating. As a result, a few minor adjustments were made to the original input model.

4. RESULTS

The PUMA GDLB (GDCS Drain Line Break) test results for safety-related issues were presented in this section. To examine important trends and the effectiveness of passive security mechanisms, critical factors were selected (Ishii, M., et al., 2006).

The break rate of flow is typically lower compared to the GDCS injection rate of flow. As a result, after the GDCS injection, the water level of the RPV rose to the Depressurization Valve (DPV) inflow elevation (4.990 m). The broken line allowed the RPV water level to overflow into the Drywell (DW) once it had crossed the break elevation. In addition to the ongoing break flow, a small quantity of vapor evaporated in the SP through the DW-to-Wetwell (WW) downward vent. The gravity-driven drain flow process is the most significant phenomenon in the GDCS injection phase. Gravity-driven drain movement was predominantly in charge of cooling the interior, so the effect of natural cooling was negligible.

Note that the comparatively high decay power was what caused the RPV water level to oscillate between the GDCS injection phase. The subcooled liquid generated by the GDCS decreased shimmering and prevented the RPV coolant from heating. As a result, the RPV ability to generate steam was reduced. The RPV pressure drop, and condensation on the wall caused the DW pressure to decrease.

The GDCS tank in Figure 4.1 displays the collapsed water level. Due to its connection to two discharge lines, of which one was damaged, the water level in this tank fell more rapidly than in other tanks. The RPV pressure was higher than the DW pressure, so the GDCS discharge paths started to inject. The gas pressure inside the GDCS tank was equal to that of the DW tank, thanks to the shielding gas line.

Figure 4.2 depicts the collapsed water level in the lower DW. The level of water did not rise during the blow-down phase because of the flickering of the liquid break flow. Eventually, the discharge of the RPV and the failure of the GDCS outflow line caused the DW level of water to rise. Throughout the long-term cooling phase, the water level in the DW decreased marginally due to the capacity contraction caused by a reduction in water density, as the temperature of the water in the lower DW rose.

The safety limit for the containment is 414 kPa (60 psia), and Figure 4.3 depicts the Upper DW pressure transient. Over the phase of blow-down, the pressure rose from 215 to 230 kPa. Consequently, the pressure decreased to 185 kPa during the GDCS injection period. Due to the comparatively high decay power, the pressure started to rise. During the lengthy cooling stage, heat transfer caused by SP, ICS, and PCCS limited the DW (drywell) pressure rise to a maximum of 270 kPa before it dropped to 250 kPa. During the long-term chilling phase, the upper DW maximum temperature was limited to 135 °C (not shown). Over the GDCS injection phase, the VB (vacuum breaker) control valves opened while the Wetwell (WW) pressure was larger than the DW (drywell) pressure. By taking this move, the PCCS will remain operational in the extended cooling period. The vacuum breaker (VB) process stops non-condensable gases from flowing backward through the PCCS vent lines from the WW to the PCCS.

The RPV steam dome pressure is depicted in Figure 4.4. Because of the break flow, steam discharge via the Automatic Depressurization System (ADS), and extremely subcooled water from the GDCS pool, the pressure swiftly decreased from 1035 kPa to

180 kPa. According to the ATHLET simulation, the ADS actuation started as soon as the break was initiated. The pressure rose to 270 kPa during the long-term cooling phase and then dropped to 250 kPa at the conclusion of the test. Also, Figure 4.5 displays the overall collapsed water level of an RPV downcomer was determined by a differential pressure transducer. Over the blow-down stage, when the water level fell from 1,820 m to 1,500 m in 130 s (intact line), the GDCS injection was visible. The Top of Active Fuel (1.623 m) was 0.123 m higher than this minimum level of 1.500 m.

The total PCCS inlet fluxes are depicted in Figure 4.6. During the first stage of the blow-down, the incoming flow increased. There was practically no flow throughout the GDCS injection period due to the reduction in RPV steam production. Once the RPV water was heated to saturation and the boiling process began, the PCCS intake flow increased. As the ICS condensers and SP water eliminated decomposition heat during the long-term chilling period, the PCCS function was deactivated. The ICS and PCCS were created to return condensation to the RPV.

In Figure 4.7, ICSI inlet fluxes are depicted. Over the first stage of the blow-down, the incoming flow increased. In the GDCS injection phase, the flow was reduced due to the reduction in RPV steam production. Throughout the prolonged chilling procedure, the ICS continued to remove heat.

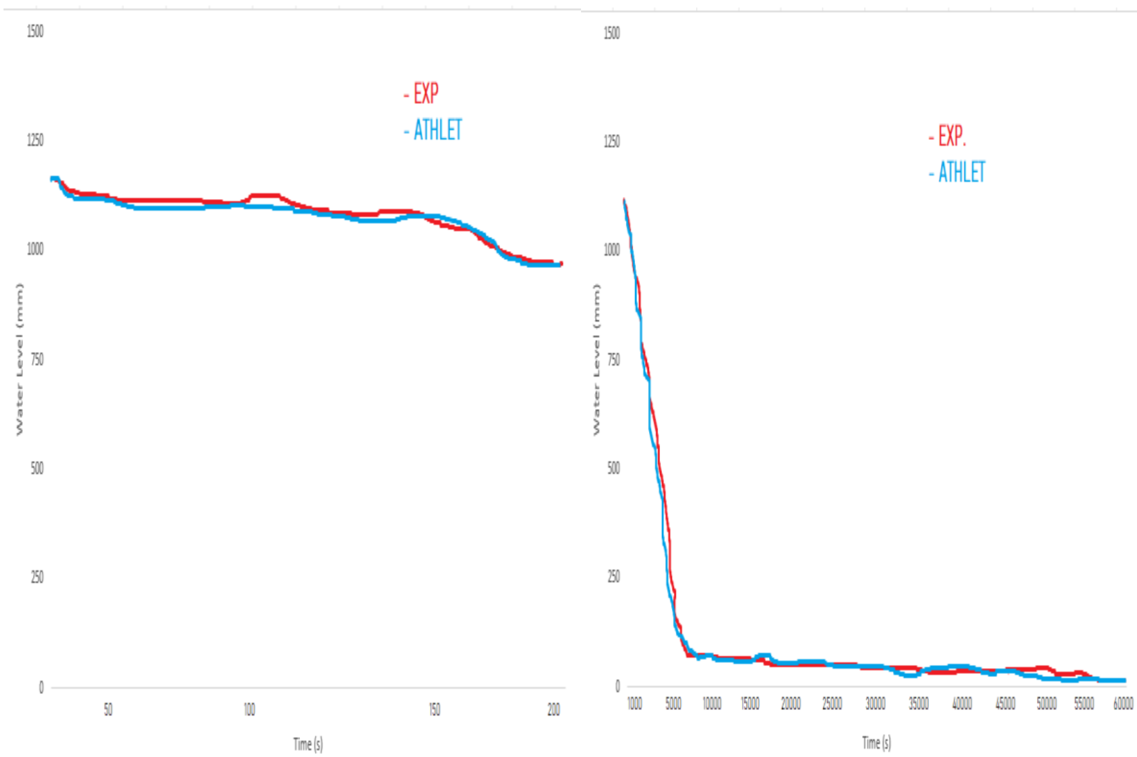
The water inventories of the RPV and GDCS pools are seen in Figures 4.8 and 4.9. Due to water injection from the GDCS pool, the RPV water supply increased. Except for the evaporation of some injected water, the water inventory loss in the GDCS was almost equal to the water inventory gain in the RPV.

Figures 4.10 and 4.11 show the temperature and pressure of the gas space in the SP. In general, steam discharge from the SRV lines linked to the SP water space causes the SP gas for the test to slightly rise during the blowdown phase. Depending on the kind of LOCA experiment, the SP gas space temperature decreases dramatically after the initial GDCS injection before increasing substantially to a temperature of 95 to 108 C at the conclusion of the testing. Similar to the SP gas temperature profile, the SP gas space pressure also declined noticeably at the start of the GDCS phase before rising to a pressure of around 260–270 kPa for this test and almost 300 kPa for the rest of the test.

Figures between 4.1 and 4.9 yielded important information with GDCS turned off. Figures between 4.10 and 4.13 show the values we get when PCCS is turned off, which is another scenario. In these figures, some values on the ATHLET were attempted by turning off the PCCS and comparing them with the experiment.

Figure 4.12 shows the RPV steam dome pressure for the LOCAs testing. From the start of the blowdown phase until the conclusion of the GDCS period, the pressure trends of the ATHLET and the experiment were comparable. During the first stage of the GDCS period, the pressure quickly decreased from a minimum of 200 kPa to a value of around 1034 kPa. Due to the diminishing core decay power output and the functioning of the passive heat removal systems, the pressure then slightly rose and stayed constant during the duration of the test at a pressure of approximately 230-270 kPa.

Figure 4.13 displays the collapsed downcomer water level in the RPV for the Small Break LOCA experiment. The loss of RPV inventory from the rupture start and the ADS function caused the RPV water levels to consistently decrease in all blowdown scenarios. The RPV water level swiftly increases in response to the GDCS operation's water replenishing during the early stages of the GDCS period. For the PCCS disabled test, the RPV water level stayed essentially constant at a certain value from the beginning of the long-term cooling phase to the conclusion of the GDCS period.



a) GDCS Water Level between 0-200 s b) GDCS Water Level between 0-
600000 s

Figure 4.1 GDCS Water Level(mm).

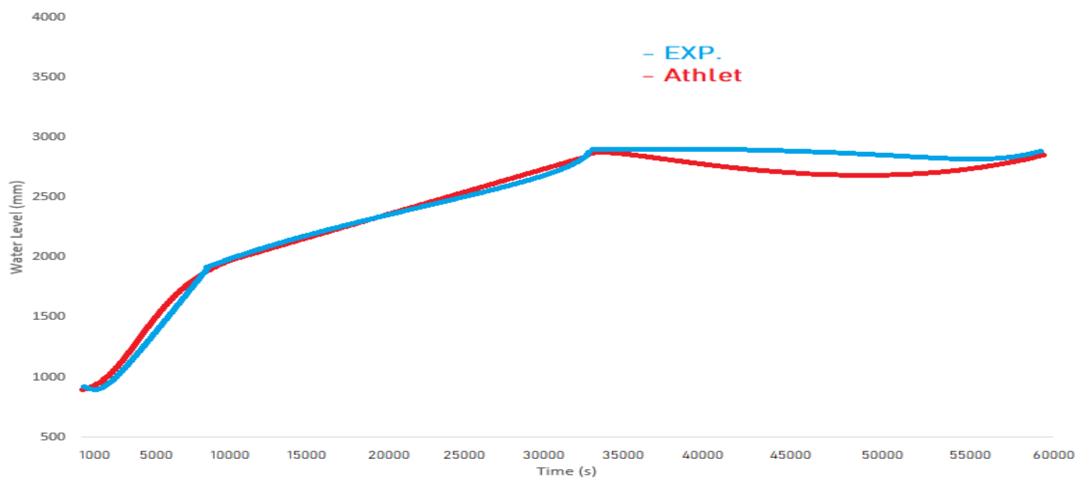


Figure 4.2 Lower Drywell Water Level(mm).

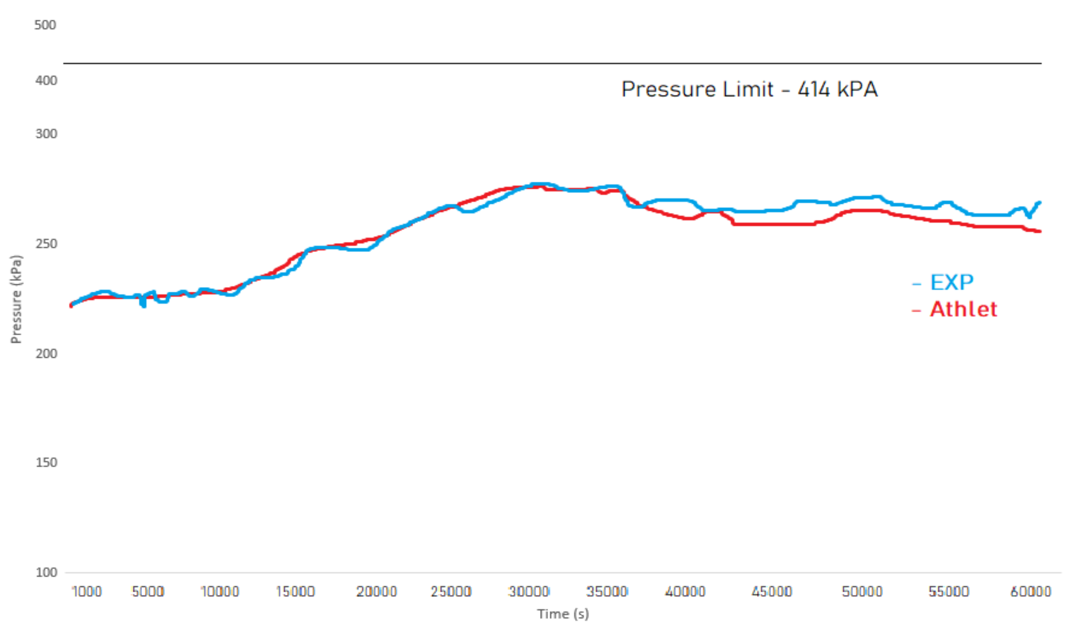


Figure 4.3 Upper Drywell Pressure (kPa).

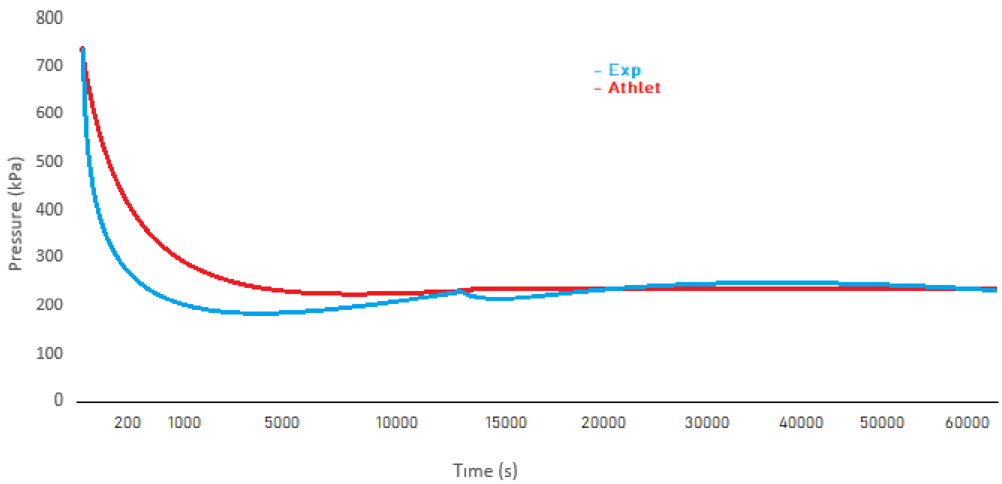


Figure 4.4 Steam Dome Pressure of RPV.

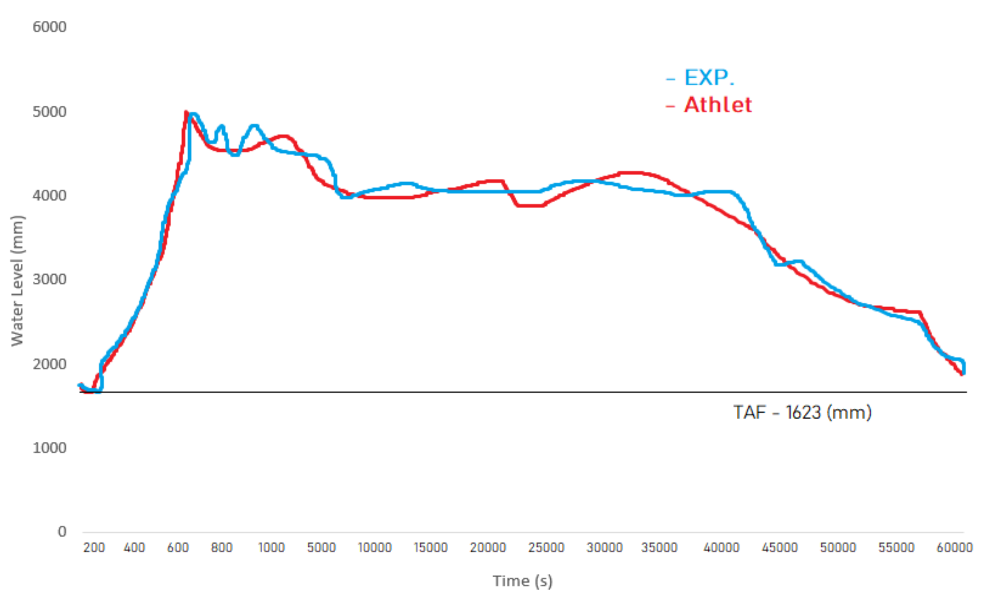


Figure 4.5 Downcomer Collapsed Water Level of RPV (mm).

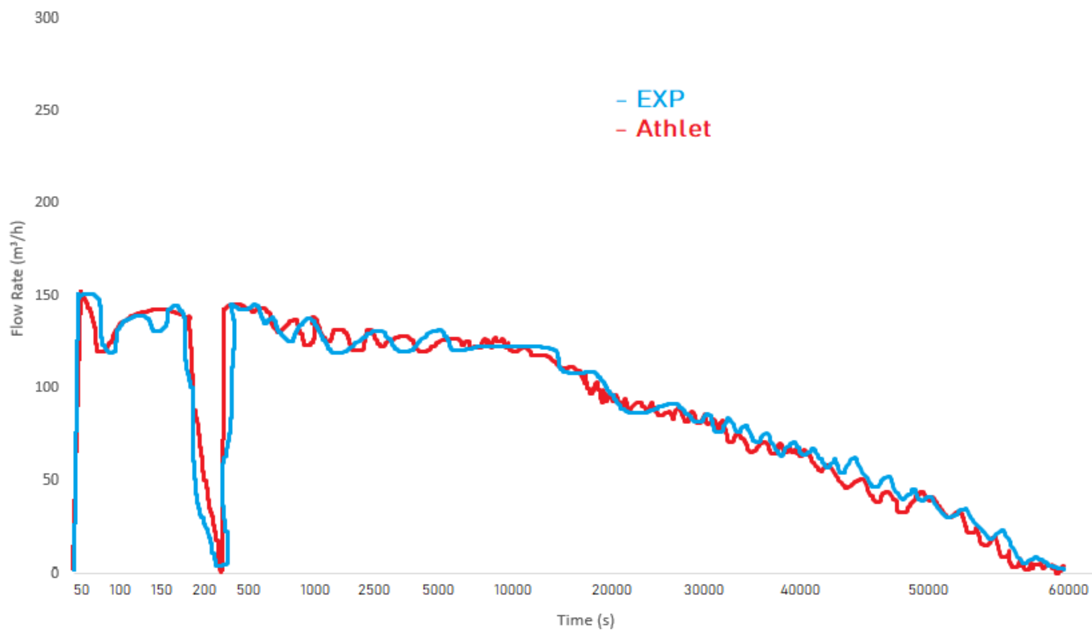


Figure 4.6 PCCS Flow Rate(m³/s).

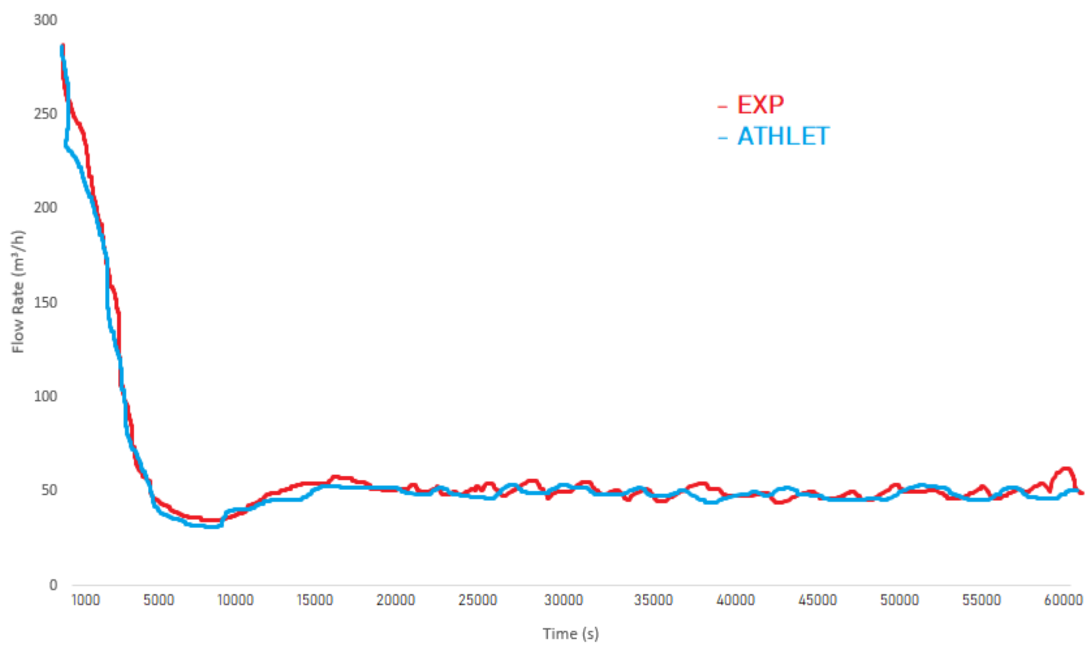


Figure 4.7 ICS Flow Rates (m³/h).

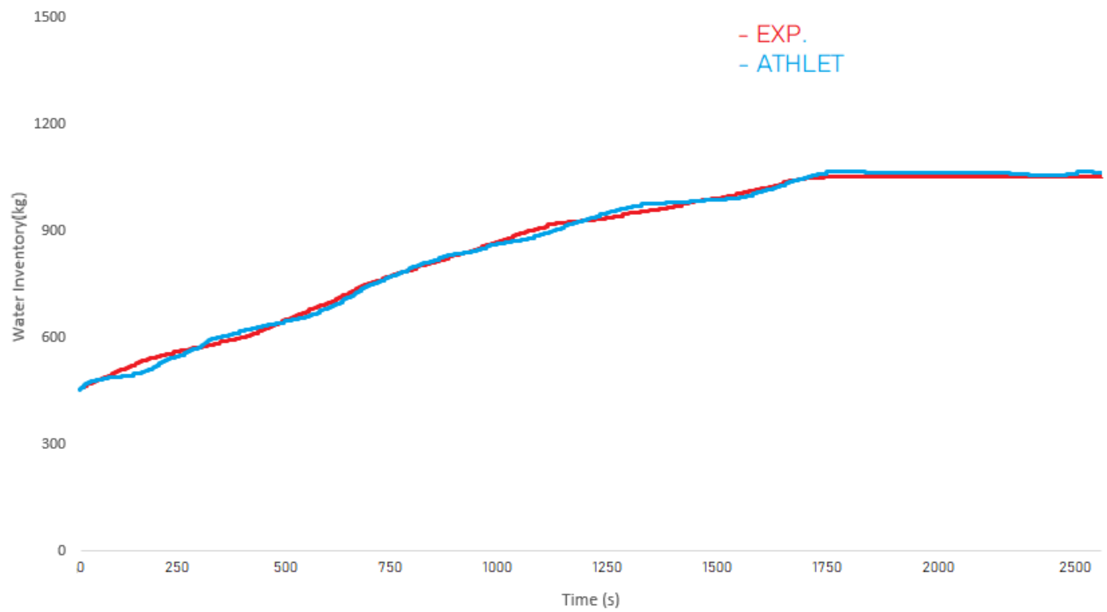


Figure 4.8 Water Inventory of RPV (kg).

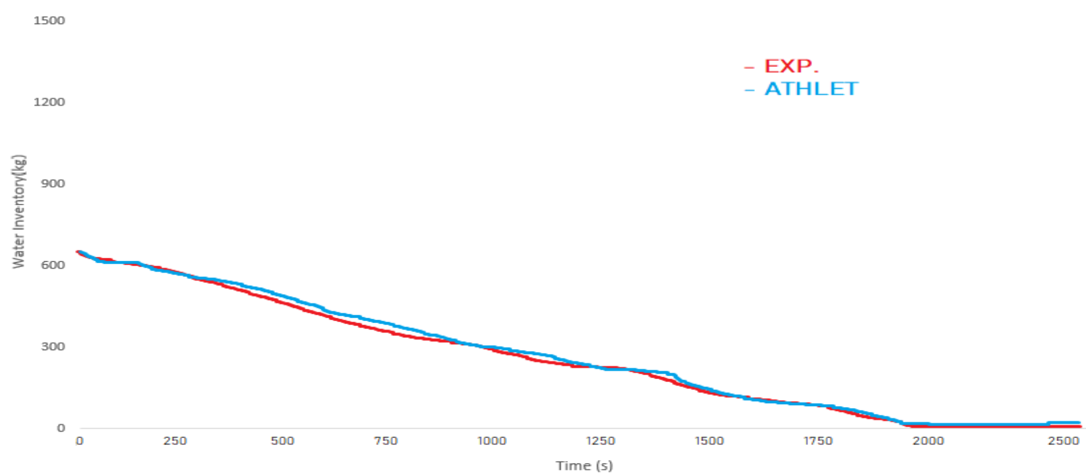


Figure 4.9 Water Inventory of GDCS (kg).

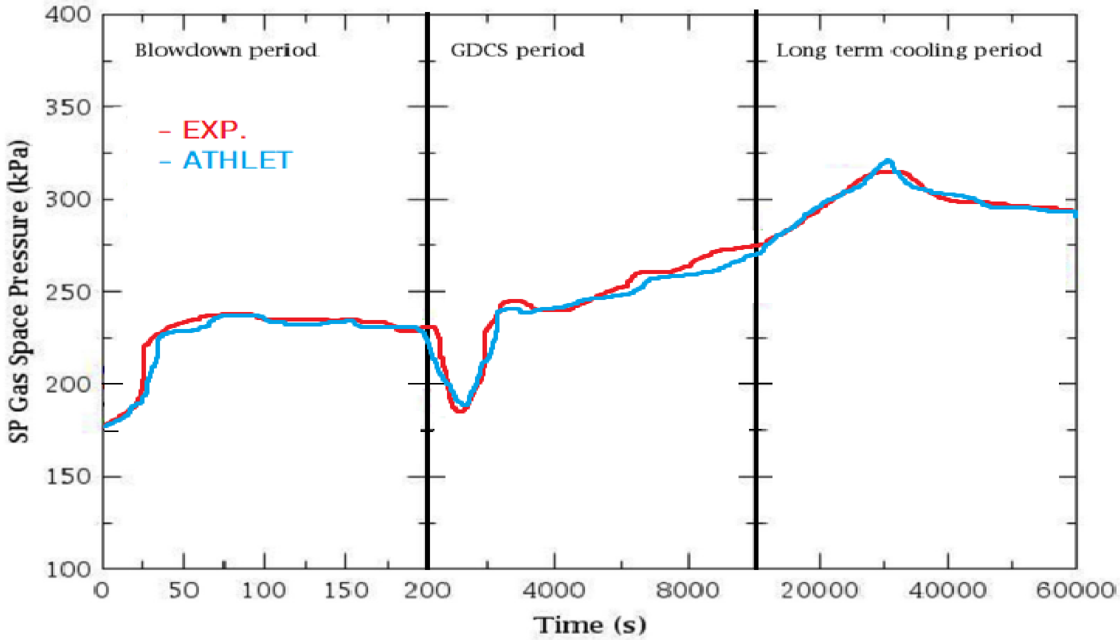


Figure 4.10 SP Gas Space Pressure (PCCS Disabled).

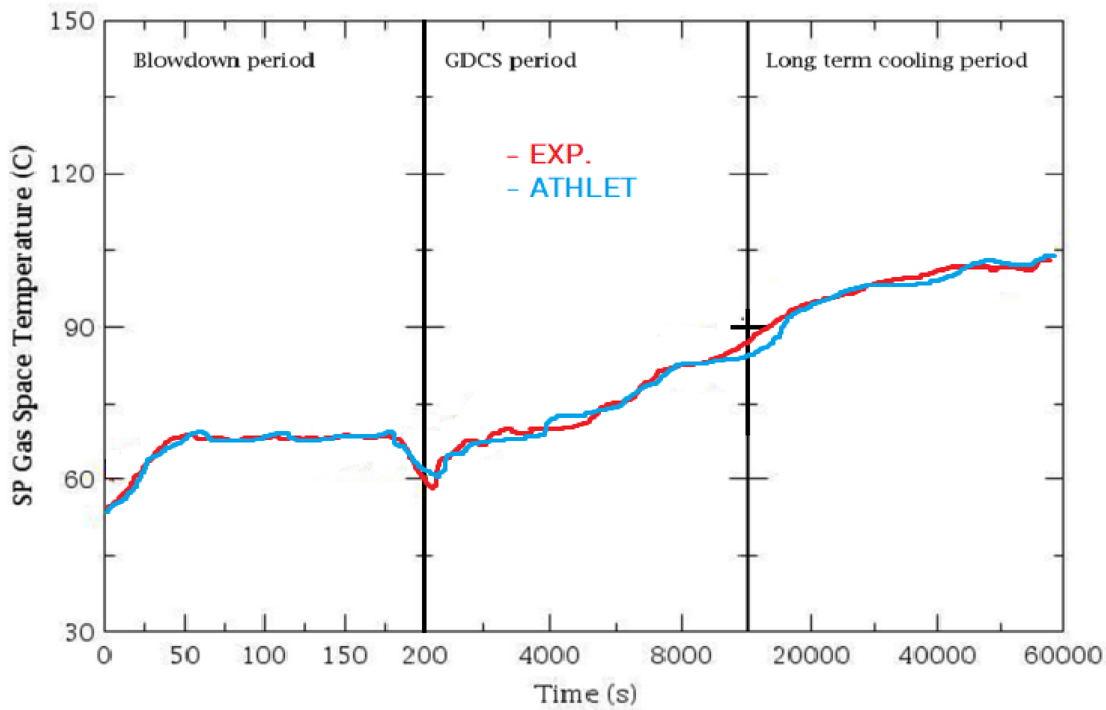


Figure 4.11 SP Gas Space Temperature (PCCS Disabled).

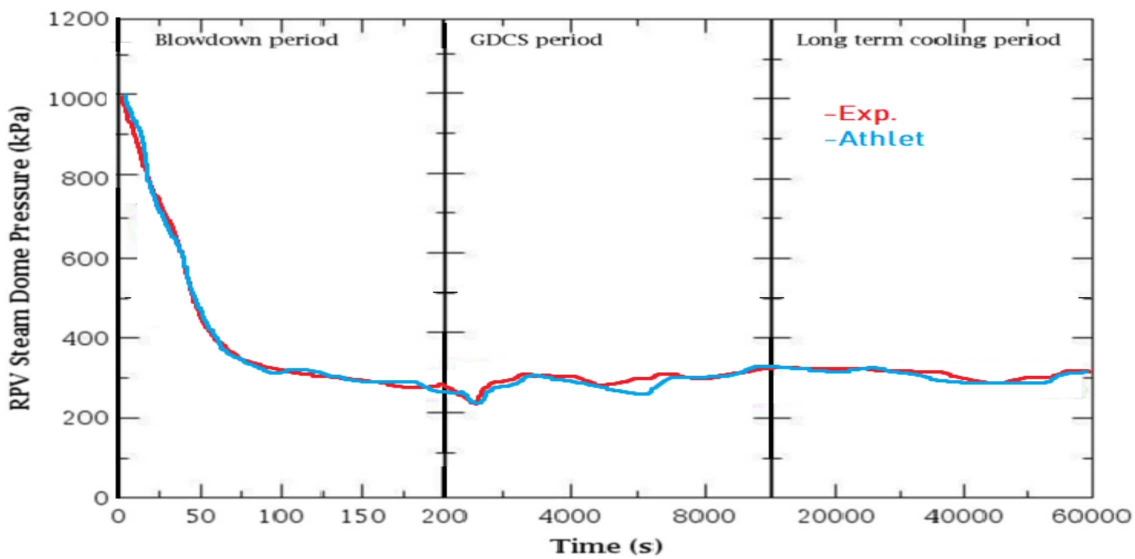


Figure 4.12 RPV Steam Dome Pressure (PCCS Disabled).

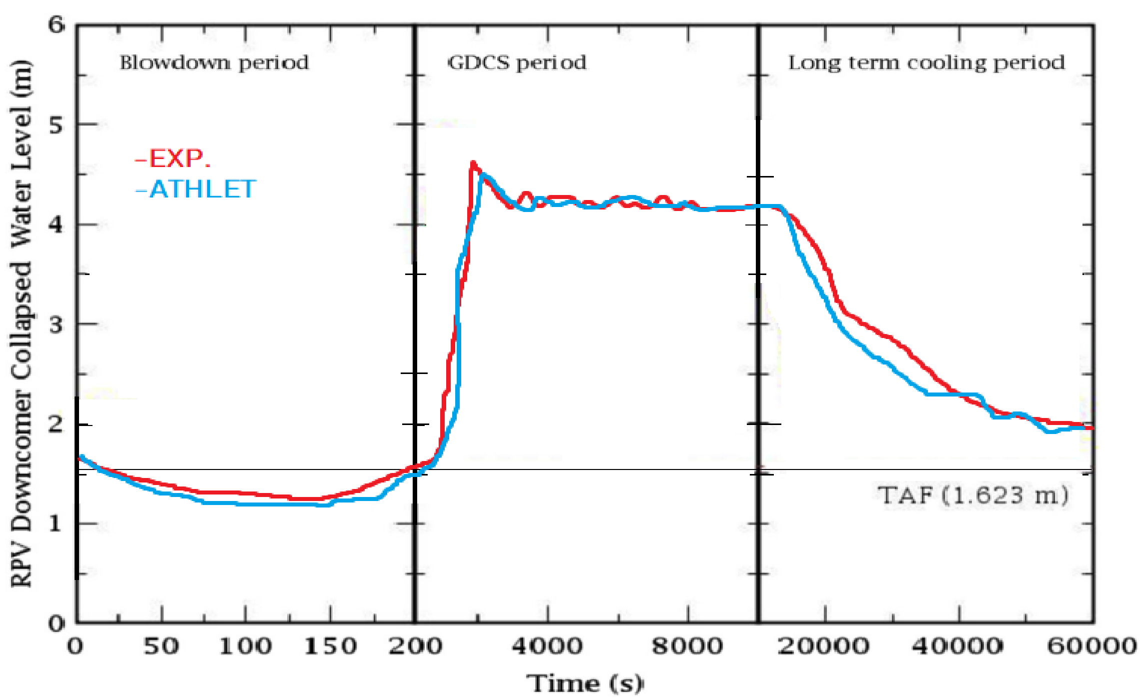


Figure 4.13 RPV Downcomer Collapsed Water Level (PCCS Disabled).

5. CONCLUSION

An extensive test was conducted at the PUMA facility, which is modeled on the ATHLET, to analyze the effectiveness of the passive safety system throughout the hypothetical GDLB (Gravity Driven Line Break) and PCCS LOCAs with the PCCS disabled. The most effective defense against the discharge of radioactive material into the environment is a nuclear power station containment. Therefore, during any fictitious event, the containment system integrity should be preserved. In this design, passive safety devices were used to keep the confinement atmosphere's pressure and temperature within predetermined limits. The development of a reliable containment cooling system is therefore one of the key components of creating an advanced reactor. In this investigation, the researchers identified explanations for the discrepancies between experimental results and theoretical predictions. Some examples of these confounding variables are:

- research assumptions,
- the geometry of the test facility's (the number of control volumes, the placement of junctions, etc.),
- the code structure,
- inaccuracies in the simulation of melt repositioning or material oxidation,
- difficulties in quenching the cylinder modeling.

The results of the integral tests demonstrated the GDCS, which served as a fallback core water injection system, delivered sufficient water to keep the RPV coolant level above the active fuel cap of 1,623 m. The two-phase water level was greater than the TAF so there was no core uncover, even though the lowest collapsed water level of 1.500 m was approximately 8% lower than the TAF (1.623 m). Throughout the long-term cooling phase, the containment system pressure was maintained below the 414 kPa design limit. The ICS and PCCS are the most significant safety parts of the system.

Based on test data and comparative findings for all break LOCA tests, it is demonstrated that the passive security mechanism of the PUMA facility typically performs out the safety tasks during the core degradation and containment elimination processes following the LOCAs break scenario. By comparing significant factors, the responses of systems that are passive to different LOCA break occurrences are illustrated and investigated.

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VITA

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