

Scaling in multiphase systems application to accidental scenarios in PWR

M. D. Domingo, A. Escrivá & J. L. Muñoz-Cobo
Universidad Politécnica de Valencia, Spain

Abstract

This paper describes the work done in collaboration with the Spanish Nuclear Regulatory Commission (CSN), and it is related with scaling analysis in multiphase systems like pressure water reactors (PWR) during accidental conditions. A method was developed in order to identify possible deviations from thermal-hydraulic similarity, or scale distortion, between two multiphase systems; and it consists of the following steps. First governing conservation equations are derived. Then, the equations are nondimensionalized with reference parameters. And an order of magnitude analysis is performed based on the numerical values of the nondimensionalized coefficients with only the large order terms being retained, which are the terms that impact on system response. So that, equations are used to identify the leading processes and quantify the scale distortions in both systems. The scaling criteria are expressed through π -Groups. This technique has been used to assess the capability of a test facility to simulate the global system response of a typical PWR during a small break loss-of-coolant accident (SBLOCA). After developing governing conservation equations, it obtained a simple multi-phase pressure rate equation useful for the analysis of the depressurization of a system containing subcooled liquid and saturated fluid, like the vessel or the hot-legs in a power plant. Then the equation is nondimensionalized. And it is found that the same nondimensional groups are important for both systems. Indicating that, although there are some distortions in scaling, the behaviour of both systems is very similar.

Keywords: SBLOCA, scaling method, nondimensional groups, distortions in scaling, governing conservation equations.



1 Introduction

The knowledge of thermal-hydraulic phenomena occurring in nuclear power plants (NPP) during an accident is very important in nuclear safety. As full-scale testing is usually impossible to perform it, then a number of test facilities have been built in order to investigate the phenomena which take place in transients or possible accidents in NPP.

Several tests carried out on these facilities allow identify relevant phenomena during transient conditions. Then, measured data are compared with simulations obtained with thermal-hydraulic codes, such as TRACE, to demonstrate their capability to reproduce experimental conditions.

In this frame, the aim of this work is to analyze if the physical phenomena observed in the experiment Test OECD/NEA ROSA1.2 project performed in Large Scale Test Facility (LSTF) are reproduced using a scale-up model.

2 Description of plants

2.1 LSTF

The Large Scale Test Facility of Japan Atomic Energy Research Institute (JAERI) is designed to simulate thermal-hydraulic phenomena peculiar to small break loss-of-coolant accidents and operational/abnormal transients of a typical Westinghouse type, four loop 3423 MW_t pressurized water reactor (figure 1).

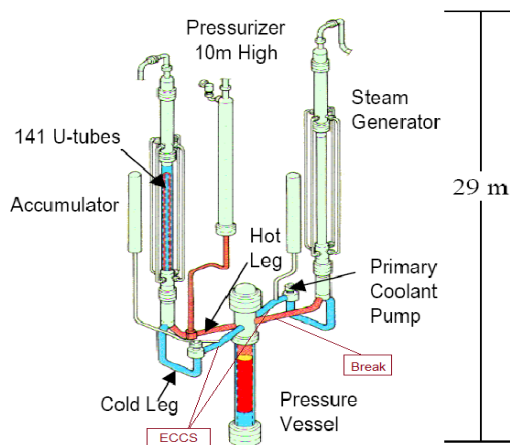


Figure 1: Large Scale Test Facility (LSTF) used to perform the ROSA test 1.2.

The LSTF is characterized by the use of prototypical-scaled components with full-height, 1/48-scaled volume and full-pressure conditions to the reference PWR. In relation with core power, it is scale by 1/48, but the maximum LSTF core power is 10 MW_t, i.e., 14% of scaled rated power of the reference PWR (JAERI-Tech [1]).

As displayed in the figure 1, the facility has two loops, A and B, with one steam generators per loop, one coolant pump per loop and one pressurizer located in loop A.

The Emergency Core Cooling system (ECCS) is formed by the accumulator tanks and the High Pressure Injection System (HPIS). The amount of water injected by the ECCS system depends on pressure. Both systems, i.e. HPIS and Accumulator Injection System (AIS), actuate when the primary pressure drops below specified pressure set points.

LSTF TRACE model contains two loops too, each one provided with primary and secondary side. The primary side comprises the pressure vessel (PV), reactor coolant pumps, cold and hot legs, loops seals, a pressurizer in loop A, the U-tubes of both steam generators and the ECCS.

2.2 3-loop PWR

A TRACE model simulating a KWU nuclear power plant has been developed (figure 2), which is a 3-loop PWR of 3010 MW_t.

Each of the three primary loops was modelled separately including hot legs, steam generators, loop seals, reactor coolant pumps, cold legs and accumulator tanks. The pressurizer and surge line are located in loop 1. The ECCS, which consists of high pressure injection system and accumulator injection system have been simulated as a boundary condition (pressure dependent).

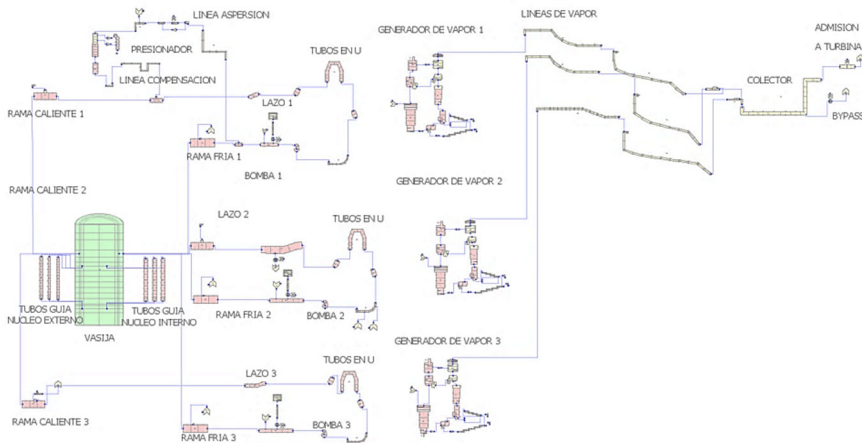


Figure 2: Three-loop PWR TRACE model.

3 Stationary initial conditions

In table 1 the stationary initial conditions (Thermohydraulic Safety Research Group [2]) are displayed for both models: test facility and 3-loop PWR.

Table 1: Steady-state condition.

	LSTF/ROSA1.2	3-loop PWR
Core Power [MW _t]	10.1	3010
Primary side		
Pressure [Pa]	15.5	15.7
Hot leg fluid temperature [K]	598.0	598.0
Cold leg fluid temperature [K]	562.0	567.0
Mass flow rate per loop [kg/s]	24.3	5287.2
Speed rotation pump [rps]	13.3	24.7
Secondary side		
Pressure [Pa]	7.3	6.9
Main feed water flow rate per loop [kg/s]	2.7	550.0
Main steam flow rate per loop [kg/s]	2.7	550.0

4 Accidental scenario in PWR: SBLOCA in a hot leg

Test ROSA1.2 consists of a hot leg SBLOCA transient assuming high pressure injection system and accumulator injection system actuation. In table 2 it is shown the sequence of major events in the transient (Muñoz-Cobo *et al.* [3]).

In this experiment, a 1% hot-leg small break LOCA is simulated in the hot-leg bottom (which corresponds to a hydraulic diameter of 10.1 mm). The transient begins with the break valve opening. The pressure in the whole system starts to fall down because of the coolant release. When the primary pressure falls below the Safety Control Rod Axe Man (SCRAM) signal set point (12.97 MPa), reactor SCRAM is initiated. After the SCRAM, the core power follows the typical curve due to the disintegration of the fission products. With reactor SCRAM, pump coastdown is initiated too; and in the secondary side, Main Steam Isolation Valves (MSIV) are closed and main feed water is terminated. The transient carries on with the Safety Injection (SI) signal when primary side pressure falls below the SI set point, and HPIS starts to work. Finally the accumulator system actuates, when the pressure is under 4.51 MPa.

Table 2: Sequence of major events during Test ROSA1.2.

Event	Condition
Hot leg break	50s
Reactor SCRAM signal	12.97 MPa
Initiation of core power decay curve simulation	With reactor SCRAM
Initiation of primary coolant pump coastdown	With reactor SCRAM
Main feedwater termination	With reactor SCRAM
Pressurizer proportional heater off	With reactor SCRAM
High Pressure Injection System (HPIS)	12.27 MPa
Accumulator Injection System (AIS)	4.51 MPa



4.1 Transient simulation

In this section it is shown a comparison of the simulation results between these TRACE models. It is provided throughout some plots (figures 3–10), which represent the main thermal-hydraulic variables: system pressures, break mass flow rates, void fraction at hot legs and mass flow rate through U-tubes in Steam Generators (SG).

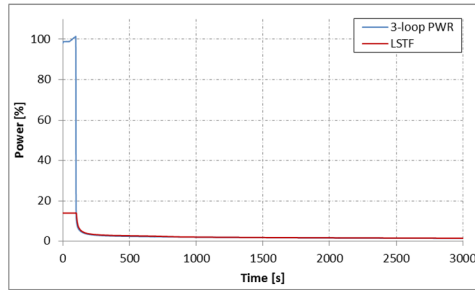


Figure 3: Decay power versus time at both plants.

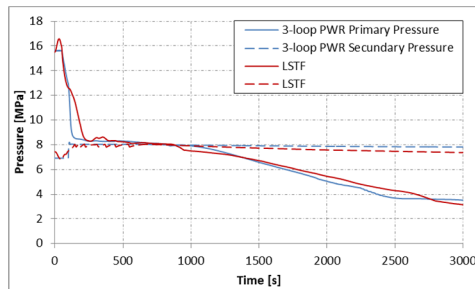


Figure 4: Simulated primary and secondary pressures at both plants.

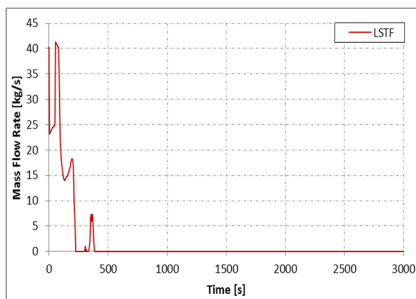


Figure 5: Mass flow rate through loop A (LSTF).

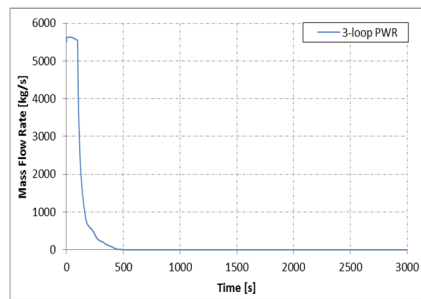


Figure 6: Mass flow rate through loop 1 (3-loop PWR).

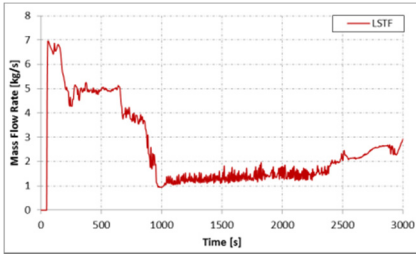


Figure 7: Computed mass flow rate through the break at LSTF.

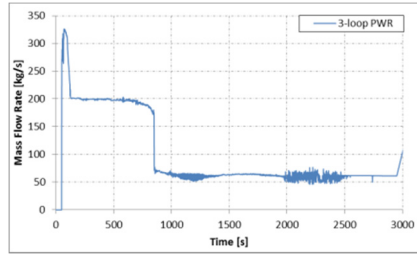


Figure 8: Computed mass flow rate through the break at 3-loop PWR.

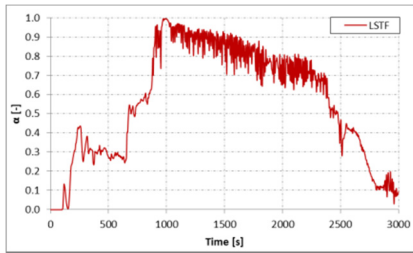


Figure 9: Void fraction at the position where the break occurs (LSTF).

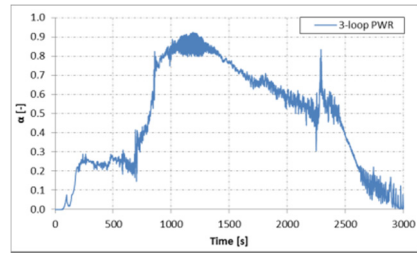


Figure 10: Void fraction at the position where the break occurs (3-loop PWR).

According to figures 9 and 10, flow regime in the hot leg changes from single phase liquid to two-phase mixture at about 100s. At time $t=300$ s the void fraction is 0.3, and at time $t=1000$ s the void fraction rises again very fast and attains a very high value close to 0.95 (primary pressure falls below secondary pressure). Then void fraction inside the system starts to decrease due to the injection of the EECS that refills the primary system with water.

5 Phase descriptions

A hot leg small break loss-of-coolant transient scenario can be broken down into distinct chronological phases (Wulff and Rohatgi [4]) with the phase boundaries base on phenomenological considerations (figure 11).

5.1 Phase 1: blowdown

This phase is characterized by a high pressure subcooled depressurization. The break flow remains in single-phase liquid throughout the blowdown period. Blowdown phase ends nearly when SCRAM signal occurs.

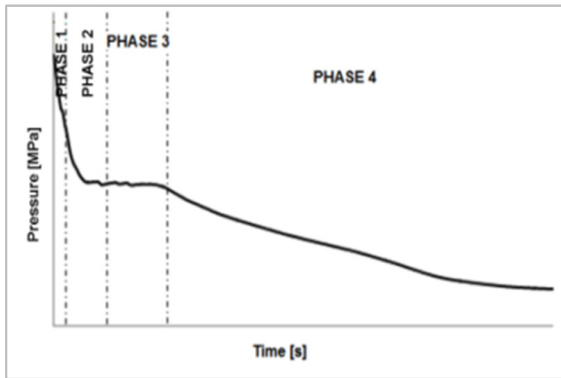


Figure 11: Transient phases.

5.2 Phase 2: natural circulation

As the reactor coolant pumps coast down, two-phase natural circulation is established in the loops of the reactor. During natural circulation the decay heat is removed by boiling in the core and condensation in the SG tubes. The natural circulation phase will continue until primary pressure falls below the steam generator secondary side pressure or the liquid mass flow at the top of the U-tubes becomes zero. Phase 2 is completed at 500s. Natural circulation phase is characterized by a null rate depressurization.

5.3 Phase 3: biphasic discharge

When natural circulation phase ends, the quality of discharged flow through the break increases. This causes the pressure in primary side falls below the secondary side pressure. Then, phase 3 starts at 500s and finishes at 1000s (more or less).

5.4 Phase 4: CORE recovery

The pressure in the reactor coolant system continues to fall while the mass inventory begins to increase. It is during this period when accumulator system starts to work. Core recovery phase ends when transient ends.

6 Scaling analysis

6.1 Blowdown

With the exception of the pressurizer, the primary coolant system contains subcooled fluid, because of this, the configuration of the system consists basically of a tank of saturated liquid and vapor (pressurizer) that is connected to a large volume of subcooled liquid (the primary system). As such, the pressurizer outflow is approximately equal to the break flow. The core is adding energy to

the system during this phase and the steam generators are effective heat sinks that balance the core energy addition. And there are not inflows to the system.

6.1.1 Governing equations and nondimensionalization

Taking into account the conservation of energy and mass for the control considered (pressurizer) it is obtained a pressure rate equation and level equation to describe the performance of the system (Mitsubishi Heavy Industries, Ltd [5] and Banerjee *et al.* [6]). In dimensionless form, equations are:

$$\begin{aligned} \frac{dp^*}{dt^*} = & -\phi_3 \sum \dot{m}_{out}^* (h_f + xh_{fg} - (\mu_f + x\mu_{fg}))^* \frac{1}{M^*} \frac{\partial p}{\partial \mu} \bigg| \\ & - \phi_4 \sum \dot{m}_{out}^* (v_f + xv_{fg})^* \frac{\mu_{fg}^*}{v_{fg}^*} \frac{1}{M^*} \frac{\partial p}{\partial \mu} \bigg|_v \\ & + \phi_5 q_{net}^* \frac{1}{M^*} \frac{\partial p}{\partial \mu} \bigg|_v \end{aligned} \quad (1)$$

$$\frac{dL^*}{dt^*} = \phi_6 \left(\frac{\dot{m}_{in}^* - \dot{m}_{out}^*}{(\rho_f^* - \rho_g^*)} - \frac{(1-L^*)}{(\rho_f^* - \rho_g^*)} \right) \frac{d\rho_g^*}{dp^*} \frac{dp^*}{dt^*} \quad (2)$$

where quantities with asterisk represent normalized variables. The definition of the variables is shown in table 3.

Table 3: Variable definitions.

h_{in} : inlet liquid enthalpy	v : specific volume
h_f : saturated liquid enthalpy	v_f : saturated liquid specific volume
h_{fg} : latent heat of vaporization	v_{fg} : specific volume on phase change
L : level	x : static quality
\dot{m}_{in} : inlet mass flow rate	μ : specific internal energy
\dot{m}_{out} : outlet mass flow rate	μ_f : saturated liquid specific internal energy
p : pressure	μ_{fg} : specific internal on phase change
q_{net} : net energy addition	ρ_f : saturated liquid density
t : time	ρ_g : saturated vapor density

The nondimensional groups in equations (1) and (2) are:

$$\phi_3 = \frac{t_0 \dot{m}_0 (h_{out} - \mu)_0}{p_0 M_0} \frac{\partial p}{\partial \mu} \bigg|_{v_0} \quad (3)$$

$$\phi_4 = \frac{t_0 \dot{m}_0 v_0 \mu_{fg_0}}{p_0 M_0 v_{fg_0}} \frac{\partial p}{\partial \mu} \bigg|_{v_0} \quad (4)$$

$$\phi_5 = \frac{t_0 q_0}{p_0 M_0} \frac{\partial p}{\partial \mu} \bigg|_{v_0} \quad (5)$$

$$\phi_6 = \frac{t_0 \dot{m}_0}{M_0} \quad (6)$$

Note that parameters with 0 represent reference values. ϕ_3 is the ratio of pressure change, due to change in specific energy flow from the mass outflows, to the reference pressure. ϕ_4 is the ratio of pressure change, due to change in specific volume of the flow, to the reference pressure. ϕ_5 is the ratio of pressure change in specific energy from heat transfer, to the reference pressure. And finally, ϕ_6 is defined as the ratio of break mass flow to the reference system mass.

6.1.2 Scaling analysis results in blowdown phase

In the depressurization equation the break volumetric flow is the dominant term (table 4).

Table 4: Nondimensional groups.

Nondimensional group	LSTF	3-loop PWR
ϕ_3	0.045	0.039
ϕ_4	0.296	0.265
ϕ_5	0.024	0.005
ϕ_6	1	1

Since facilities are working at the same conditions there is a very good agreement for the ϕ_4 group (table 5). Order of magnitude analysis shows that the effects of pressurizer heaters are an order of magnitude smaller than break volumetric flow so that it is not important the distortion scale between both plants for ϕ_5 parameter.

Table 5: Comparison between LSTF and 3-loop PWR.

$\frac{\phi_{3,LSTF}}{\phi_{3,3-loopPWR}}$	1.15
$\frac{\phi_{4,LSTF}}{\phi_{4,3-loopPWR}}$	1.12
$\frac{\phi_{5,LSTF}}{\phi_{5,3-loopPWR}}$	4.61
$\frac{\phi_{6,LSTF}}{\phi_{6,3-loopPWR}}$	1.00

6.2 Natural circulation

As the reactor coolant pumps coast down, two-phase natural circulation is established in the loops of the reactor. Steam begins to appear in the primary system.

6.2.1 Governing equations and nondimensionalization

As the previous phase, natural circulation phase is evaluated from the viewpoint of mass and energy (Banerjee *et al.* [6]). And finally, it is obtained a pressure rate equation to describe the performance of the system. In dimensionless form, equation is:

$$\begin{aligned} \frac{dp^*}{dt^*} = & \Psi_1 C_{2,l}^* I_{a,l}^* + \Psi_2 C_{2,l}^* I_{b,l}^* + \Psi_3 C_{2,l}^* I_{c,l}^* \\ & + \Psi_4 C_{2,m}^* I_{a,m}^* + \Psi_5 C_{2,m}^* I_{b,m}^* + \Psi_6 C_{2,m}^* I_{c,m}^* \\ & + \Psi_7 C_{2,N}^* I_{a,N}^* + \Psi_8 C_{2,N}^* I_{b,N}^* + \Psi_9 C_{2,N}^* I_{c,N}^* \\ & + \Psi_{10} C_1^* \Pi_l^* + \Psi_{11} C_1^* \Pi_m^* + \Psi_{12} C_1^* \Pi_N^* \end{aligned} \quad (7)$$

where it is considered subscript l denotes subcooled liquid, m denotes a homogenous steam-water mixture and N denotes steam or nitrogen.

Coefficients I_k , Π_k , are defined for convenience; using k to denote either field l, m and N.

$$I_k = \sum \dot{m}_{in,k} (h_{in,k} - \mu_k) - \sum \dot{m}_{out,k} (h_{out,k} - \mu_k) + \dot{q}_{net,k} \quad (8)$$

$$\Pi_k = v_k \left[\sum \dot{m}_{in,k} - \dot{m}_{out,k} \right] \quad (9)$$

The nondimensional groups in equation (7) are:

$$\Psi_1 = C_{2,l,0} \dot{m}_0 (h_{in,l} - \mu_l)^{t_0} / p_0 \quad (10)$$

$$\Psi_2 = C_{2,l,0} \dot{m}_0 (h_{out,l} - \mu_l)^{t_0} / p_0 \quad (11)$$

$$\Psi_3 = C_{2,l,0} q_0 t_0 / p_0 \quad (12)$$

$$\Psi_4 = C_{2,m,0} \dot{m}_0 (h_{in,m} - \mu_m)^{t_0} / p_0 \quad (13)$$

$$\Psi_5 = C_{2,m,0} \dot{m}_0 (h_{out,m} - \mu_m)^{t_0} / p_0 \quad (14)$$

$$\Psi_6 = C_{2,m,0} q_0 t_0 / p_0 \quad (15)$$

$$\Psi_7 = C_{2,N,0} \dot{m}_0 (h_{in,N} - \mu_N)^{t_0} / p_0 \quad (16)$$

$$\Psi_8 = C_{2,N,0} \dot{m}_0 (h_{out,N} - \mu_N)^{t_0} / p_0 \quad (17)$$

$$\Psi_9 = C_{2,N,0} q_0 t_0 / p_0 \quad (18)$$

$$\Psi_{10} = C_{1,0} v_{l,0} \dot{m}_0 / p_0 \quad (19)$$

$$\Psi_{11} = C_{1,0} v_{m,0} \dot{m}_0 / p_0 \quad (20)$$

$$\Psi_{12} = C_{1,0} v_{N,0} \dot{m}_0 / p_0 \quad (21)$$

where it has been defined C coefficients as follows:

$$C_1 = \frac{1}{\frac{\rho_l V_l}{\partial p / \partial v_l|_{\mu_l}} + \frac{\rho_m V_m}{\partial p / \partial v_m|_{\mu_m}} + \frac{\rho_N V_N}{\partial p / \partial v_N|_{\mu_N}}} = \frac{1}{\sum_k \frac{\rho_k V_k}{\partial p / \partial v_k|_{\mu_k}}} \quad (22)$$

$$C_{2,k} = \frac{\frac{\partial P}{\partial \mu_k}|_{v_k} / \frac{\partial P}{\partial v_k}|_{\mu_k}}{\sum_k \frac{\rho_k V_k}{\partial p / \partial v_k|_{\mu_k}}} \quad (23)$$

7 Conclusions

It has been obtained a model to simulate test ROSA1.2 in a typical 3-loop PWR with TRACE code; getting quite acceptable results, at least for the most important processes.

Then, transient has been divided into several phases, identifying the most relevant phenomena in order to perform the scaling analysis between test facility and a typical 3-loop PWR.

And finally, it has been proposed a scaling methodology for single-phase and multi-phase volumes. The behavior of the system has been characterized using nondimensional equations to examine the scalability quantitatively.

Initial results show that the controlling mechanism for the depressurization is almost identical for both plants in the blowdown phase. There is a scaling distortion but it does not impact on the depressurization behavior.

Now it is needed to finish the calculations of the nondimensional groups for the remaining phases and thus to compare both plants; taking into account that the most general case is contemplated in natural circulation phase (equations from (10) to (22)).

Acknowledgement

With thanks to the Spanish Nuclear Regulatory Commission (CSN) for funding and supporting.

References

- [1] JAERI-Tech, "ROSA-V Large Scale Test Facility (LSTF) system description for the third and fourth simulated fuel assemblies", 2003.



- [2] Thermohydraulic Safety Research Group, “Final Data Report of OECD/NEA ROSA Project Test 1-2 (1% hot leg break LOCA experimental with HPI: SB-HL17 in JAEA)”, Japan Atomic Energy Agency, 2008.
- [3] Muñoz-Cobo J. L., Romero A. & Chiva S. “Analysis with TRACE code of ROSA Test 1.2: Small LOCA in the Hot-Leg with HPI and Accumulator actuation”, NUREG/IA-420, 2012.
- [4] Wulff W. & Rohatgi U. S. “System Scaling For The Westinghouse AP600 Pressurized Water Reactor and Related Test Facilities”, NUREG/CR-5541, 1998.
- [5] Mitsubishi Heavy Industries, Ltd, “Scaling Analysis for US-APWR Small Break LOCAs”, 2009.
- [6] Banerjee S., Ortiz M. G., Larson T. K. & Reeder D. L., “Top-Down Scaling Analyses Methodology for AP600 Integral Tests”, INEL-96/0040, May 1997.

