# ANALYSIS OF MEASURED AND CALCULATED COUNTERPART TEST DATA IN PWR AND VVER 1000 SIMULATORS

by

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**This paper presents an overview of the "scaling strategy", in particular the role played by the counterpart test methodology. The recent studies dealing with a scaling analysis in light water reactor with special regard to the VVER 1000 Russian reactor type are presented to demonstrate the phenomena important for scaling. The adopted scaling approach is based on the selection of a few characteristic parameters chosen by taking into account their relevance in the behavior of the transient. The adopted computer code used is RELAP5/Mod3.3 and its accuracy has been demonstrated by qualitative and quantitative evaluation.**

**Comparing experimental data, it was found that the investigated facilities showed similar behavior concerning the time trends, and that the same thermal hydraulic phenomena on a qualitative level could be predicted. The main results are: PSB and LOBI main parameters have similar trends. This fact is the confirmation of the validity of the adopted scaling approach and it shows that PWR and VVER reactor type behavior is very similar. No new phenomena occurred during the counterpart test, despite the fact that the two facilities had a different layout, and the already known phenomena were predicted correctly by the code. The code capability and accuracy are scale-independent. Both characteristics are necessary to permit the full scale calculation with the aim of nuclear power plant behavior prediction.**

*Key words: nuclear reactor safety, scaling analysis, VVER reactor, RELAP5*

## INTRODUCTION

The execution of experiments in integral test facilities (ITF) simulating the behavior of a nuclear power plant plays an important role regarding safety aspects (*i. e.*, evaluation of the safety margin) and code applications (*i. e*,. code validation and its accuracy evaluation). For considering both the system code assessment and the possibility to identify and characterize the relevant phenomena during off-normal conditions, the use of ITF is unavoidable in order to collect the experimental data because of the impossibility to perform tests at the full scale.

In the framework of these activities, Pisa University is involved in a technical assistance community independent state (TACIS) project with the EU [1] and in some OECD projects. The EU founded project consists of two different parts that are linked under the nuclear reactor safety umbrella, in particular within the framework of the accident analysis of nuclear power plants (NPP); the OECD project which has provided the experimental data is named OECD PSB-VVER. In order to use the measured data and to extrapolate them for the NPP prediction, the scaling issue (*i. e.*, the demonstration of a similarity between experiments performed in differently scaled facilities and between measured phenomena and phenomena expected in the reference NPP) plays a role of major importance.

The present work is focused on the application of relevant steps of the scaling analysis with main reference to the counterpart test (CT) methodology [2]

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from which a general approach of solving scaling problems can be extrapolated. The CT taken as an example is essentially a small break LOCA (SBLOCA) and it has been designed deriving the boundary and the initial condition from the same test performed in the loop for off-normal behavior investigation (LOBI) (that reproduced a PWR). Such a test has been performed in the PSB-VVER facility, a full height full pressure rig with a scaling factor of 1:300. The break simulator is located in the cold leg of the loop no. 4 (the loop with the pressurizer), downstream the main circulation pump; the intervention of the high pressure injection systems is not foreseen, while the low pressure injection systems are activated on a high rod temperature signal. The design of this test has been made in collaboration with EREC (Electrogorsk Research Engineering Center), following the methodology proposed by University of Pisa. This scenario has been chosen because it had already been used as a CT in the past for other experimental facilities that simulated PWR, namely for SPES (Simulatore PWR per Esperienze di Sicurezza), BETHSY (Boucle d'Études Thermal-Hydrauliques Systemes), and LSTF (Large Scale Test Facility).

The computational analysis of the CT using RELAP5/Mod.3.3 system code was possible thanks to the availability of a qualified nodalisation of the PSB facility, developed and qualified at University of Pisa. A qualitative and a quantitative evaluations of the obtained results have also been reported, the last step foreseeing the use of the fast Fourier transform (FFT) based method.

## PSB-VVER FACILITY DESCRIPTION

The PSB-VVER is a full height integral test facility, extensively described in [3] and shown in fig. 1; power and volume are scaled at 1:300. The facility has four loops (each one consists of a hot leg, a steam generator, a loop seal, a main circulation pump and a cold leg), a pressurizer connected via the surge line to the hot leg of the loop 4, the emergency core cooling system (ECCS) which is pro-

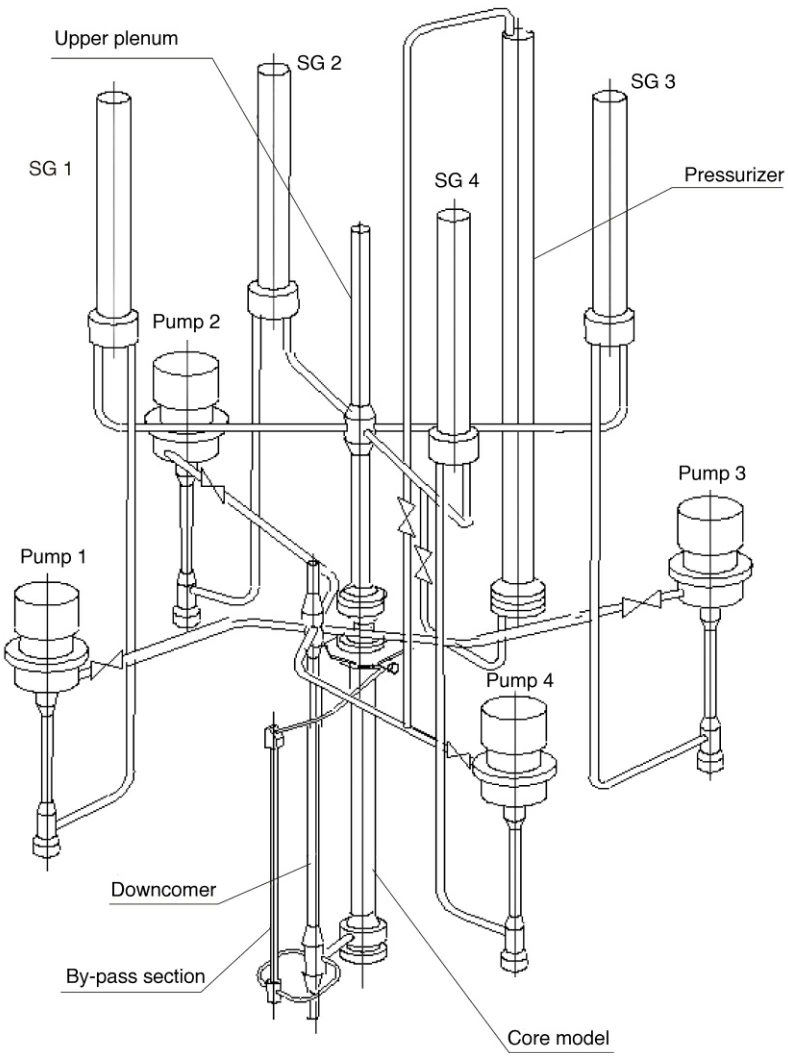


**Figure 1. PSB-VVER: general view of the facility**

vided by an active pump simulating high and low pressure injection systems, and four hydro-accumulators. All system components are insulated from the environment with glass wool to limit the heat losses.

The main parts of the VVER vessel are reproduced in the facility by separate pipes: one for the downcomer, one for the core model and upper plenum, and one for the core by-pass. A horizontal pipe connects the downcomer to the lower plenum. Another by-pass links the downcomer to the upper plenum.

The core model contains 168 fuel rod simulators with a uniform power profile and a central unheated rod. The active bundle is of electrical type and has a hexagonal cross section. Up to now, the core power has been limited to the maximum of 1.5 MW (15% of the nominal scaled power). A new core is foreseen with the power up to 10 MW (the right scaled value). The by-pass section is heated via the same elevation range of the core, to simulate the heating that water receives in the channels within the reactor core, in which the coolant flows from the lower plenum to the upper plenum, by-passing the assemblies.

The primary side of the steam generator consists of a hot and a cold collector and of 34 tubes coiled in 10 complete turns with 51 mm difference from the inlet and outlet height. The length of one tube is the same as the one of the reference plant. The distributor of the feed water is a ring with several holes placed above the steam generator tubes. Separators are completely absent. The four steam generators are connected to a common steam header via a "small power" steam line.

A comparison between VVER 1000 NPP and PSB-VVER test facility main data [4] is presented in tab. 1.

## THE SCALING ISSUE

### Addressing the scaling issue

Reactor safety integral test facilities are normally designed to preserve the geometrical similarity with the reference reactor system. Generally, all main components (*e. g.* rector pressure vessel, downcomer, rod bundle, loop piping, *etc*.) and the engineered safety system (high pressure injection system – HPIS, low pressure injection system – LPIS, accumulators, auxiliary feed water, *etc*.) are presented. ITFs are used to investigate, by direct simulation, the behavior of a NPP in cases of the off-normal or accident conditions.

The term scaling is in general understood in a broad sense covering all differences existing between a real full size plant and the corresponding ex-

**Table 1. Main design parameters of PSB-VVER ITF, compared with those of VVER-1000 NPP**

| Name | VVER-1000 | PSB-VVER | Scale factor |
|---|---|---|---|
| Numbers of loops | 4 | 4 | – |
| Heat losses, [%] | 0.063 | 1.8 | – |
| Heat power, [MW] | 3000 | 10* | 1:300 |
| Primary circuit volume, [$m^3$] | 370 | 1.23 | 1:300 |
| Primary circuit pressure, [MPa] | 15.7 | 15.7 | 1:1 |
| Secondary circuit pressure, [MPa] | 6.3 | 6.3 | 1:1 |
| Coolant temperature, [K] | 563/593 | 563/593 | 1:1 |
| Core lengrh, [m] | 3.53 | 3.53 | 1:1 |
| Number of fuel rods | 50856 | 169 | 1:300 |
| Core volume, [$m^3$] | 14.8 | $4.9\ 10^{-2}$ | 1:302 |
| Upper plenum volume, [$m^3$] | 61.2 | $20.0\ 10^{-2}$ | 1:306 |
| Downcomer volume, [$m^3$] | 34.0 | $11.0\ 10^{-2}$ | 1:309 |
| Hot legs volume, [$m^3$] | 22.8 | $8.0\ 10^{-2}$ | 1:285 |
| Cold legs volume, [$m^3$] | 60.0 | $24.0\ 10^{-2}$ | 1:250 |
| Number of steam generators | 4 | 4 | – |
| Heat exchanging surface, [$m^2$] | 6115 | 18.2 | 1:336 |
| Water volume in steam generator primary circuit, [$m^3$] | 21.0 | $6.8\ 10^{-2}$ | 1:309 |
| Pressurizer volume, [$m^3$] | 79 | $26.3\ 10^{-2}$ | 1:300 |
| Number of hydro accumulators | 4 | 4 | – |
| Number of pumps | 4 | 4 | – |
| Volume of hydro accumulators, [$m^3$] | 240 | $80\ 10^{-2}$ | 1:300 |
| Water volume in accumulators, [$m^3$] | 200 | $66.6\ 10^{-2}$ | 1:300 |

* for the new core

perimental facility. An experimental rig may be characterized by geometrical dimension and shape, arrangement and availability of components, or the mode of operation (*e. g.* nuclear *vs.* electrical heating). All these differences have the potential to distort an experimental observation, precluding its direct application for the design or operation of the reference plant. Distortion is defined as a partial or total suppression of physical phenomena caused by only changing the size (geometric dimension) or the shape (arrangement of components) of the test rigs [5].

Due to the impossibility to perform relevant experiments at the full scale (full power, pressure and geometry), the use of ITF or separate effect test facility (SETF) is necessary. In order to address the scaling issue, different approaches are historically followed [2]:

(1) One tries to preserve selected non-dimensional parameters adopting the Buckingham theorem derived from the fluid balance equations,
(2) One tries to preserve selected non-dimensional parameters adopting the Buckingham theorem derived from the semi-empirical mechanicistic equations instead of the fluid balance equations,
(3) One performs experiments at different scales, and
(4) One develops, qualifies and applies codes showing their capabilities at different scales.

The scaling approach is applicable at the level of the macro scale, component scale and micro scale.

The target of the macro scale is to evaluate the global system performance considering it as a whole (*e. g.* the prediction of the pressure behavior in case of a blow-down following a LOCA); in the component scale, the behavior of the single component is taken into account (*e. g.* downcomer-cold leg mixing in case of ECCS injection). It must be noted that at this level the design of some components adopts the most desirable scaling factor: 1:1; this occurs for the active length, for the fuel rod diameter, diameter and length of the steam generator (SG) tubes, *etc*. Finally, when the investigation is pushed at the micro scale the interest is focused on the local evolution of thermal hydraulic phenomena, such as critical heat flux occurrence in fuel rods, two phase critical flow in a break, *etc*.

The scaling analysis has mainly three objectives:
– to prove the capability of simulating an assigned phenomenon,
– to design a test facility, and
– to prove the capability of a thermal hydraulic model or of a thermal hydraulic system code.

For the test facility design three types of scaling principles have been identified in literature:

(1) Time reducing scaling: rigorous reduction of any linear dimension of the test rig would result in a direct proportional reduction in time scaling. This is considered to be of advantage only for cases where body forces due to gravity acceleration are negligible compared to the local pressure differentials.
(2) Time preserving scale: based on a scale reduction of the volume of the loop system combined with a direct proportional scaling of energy sources and sinks (core power to system volume ratio = const.).
(3) Idealized time preserving modeling procedures: based on the equivalency of the mathematical representation of the full size plant and of the test rig, it is deducted from a separated treatment of the conservation equations for all involved volume modes and flow paths assuming homogeneous fluid.

The geometrical similarity of the hardware of the loop systems has been abandoned in favor of the preservation of geometric elevations, which are decisive parameters in the case of gravity dominated and natural circulation processes. Thus the reduction of the primary system volume is largely achieved by an equivalent reduction in vertical flow cross sections.

The proof of the capability of a thermal hydraulic system code to predict all the phenomena occurring in any experiment performed at an ITF is equivalent to demonstrating the "scaling-independence" of the code. Once this condition is reached, the tested code is capable to reproduce the NPP behavior as well.

The scientific community has produced hundreds of pages of literature in the attempt to reach a solution to the scaling analysis.

The derivation of non-dimensional parameters by the direct use of balance equations gives as result more than 200 scaling factors that, as the authors say, "cannot completely satisfy all the scaling requirements" [6].

The derivation of non-dimensional parameters by the combination of balance equations and engineering judgment for emphasizing the relevant phenomena results in about 30 scaling factors, that applied to different facilities give spread values.

Summarizing, the actual state of the art in approaching the scaling issue is:
– identification and characterization of main thermal hydraulic phenomena,
– writing equations (fluid balance & mechanistic) at the macro-scale, component-scale, and micro-scale levels and deriving suitable non-dimensional parameters, and
– achieving "qualified" functional relationships between the phenomena and scale.

*Lesson learned*: the attempt to scale up all thermal hydraulic phenomena that occur during an assigned transient leads to a myriad of factors which have counterfeiting validity [6]. Therefore, as recognized by Zuber [7], an overall strategy is needed and a hierarchy in scaling factors is necessary.

The preferred scaling strategy of the University of Pisa adopts the combination of the derivation of non-dimensional parameters from balance equations and the use of system codes. In more detail it implies:

- selection of a scaling approach at the system level (macro scale): the full pressure / full height / time preserving scaling (this requires "full" bundle active length and preserving linear power. Not all ITF & SETF are suitable for scaling studies),
- considering the rod surface temperature at the hot spot as the reference thermal hydraulic parameter at the micro scale level: this assumption is guided by the finalization of the scaling analysis to the NPP safety,
- identification and creation of a hierarchy of thermal hydraulic phenomena: the adherence to the Committee on the safety nuclear installations (CSNI) lists for SETF and ITF and the evaluation of individual phenomena is strictly recommended,
- (checking of the) design of ITF,
- (checking of the) design of "counterpart experiments",
- analysis of CT experimental data: identification and explanation of detected discrepancies among corresponding values,
- application of the best estimate codes: (a) to demonstrate that discrepancies between measured and calculated trends only depend on boundary and initial condition (BIC) values (within the assigned variation ranges), and calculation accuracy is not affected by the scale of concerned ITF, (b) to perform volume scaled factor ($Kv$) calculation and explanation of discrepancies (if any) between NPP calculated and ITF measured trends considering BIC values and hardware differences, and
- connection of the uncertainty evaluation to the scaling issue: extrapolation of the error of the code in NPP prediction based on error in ITF prediction [8–10].

It could happen that some local events are not predicted because they are driven by parameters that do not appear in the balance equations, but, thanks to the correct selection of the parameters, it is possible to demonstrate that these phenomena are effectively local, that they have short duration if compared to the entire transient, and that they cannot affect the overall behavior of the main thermal hydraulic selected parameters chosen to describe the transient.

### Solving the scaling issue

As previously said, the SBLOCA performed in the PSB facility is a CT of the same test carried out in LOBI [6]. In order to define the initial and boundary conditions of the test some driving parameters have been investigated. This investigation has brought to consideration of the primary side volume (without pressurizer) ratio between LOBI and PSB as the main parameter.

The definition of a set of parameters concerning the test design was proposed by University of Pisa. Some of these parameters are: break area over primary side volume ($A_R/V_{PS}$), mass delivered from the safety injection tank (SIT) over primary side volume ($M_{SIT}/V_{PS}$), and LPIS flow rate over primary side volume ($G_{LPIS}/V_{PS}$). The approach is to assume the values adopted in the experiment performed in the LOBI facility as a reference, deriving the PSB corresponding values in the same way. After several exchanges with EREC [12, 13], this approach has been accepted and followed. The selected parameters are shown in tab. 2 [12, 14], together with the corresponding values in other ITF.

## THE CT RESULTS

### PSB SBLOCA test description

The same counterpart test experiment has been carried out in LOBI, SPES, BETHSY, and LSTF facilities, the design objective being the simulation of PWR system hydraulic response following a postulated SBLOCA accident. The break simulated was a 0.152 m (side oriented) cold leg break in the corresponding reference PWR. The HPIS intervention was not foreseen.

Starting from those past experiments, the SBLOCA test has been designed in collaboration with EREC, following the scaling law for deriving boundary and initial conditions as well as possible in order to carry out a correct counterpart test [12].

The relative scaling factors were based on the following characteristics of the facilities:

- nominal core power,
- the facility's volumetric (power) scaling factor,
- primary mass inventory (total primary mass and primary mass without the pressurizer and surge line mass),
- primary system volume (total primary volume and primary volume without the pressurizer and surge line), and
- break location.

Due to the availability of detailed LOBI experimental data and configuration, EREC chose those conditions as guidelines. Adopting the factor defined as the ratio between the primary side volume of the fa-

**Table 2. Comparison between experimental conditions CT test criteria (low power only)**

| | Unit | LOBI | PSB-VVER | SPES | BETHSY | LSTF |
|---|---|---|---|---|---|---|
| $A_R/V_{PS}$ | m$^{-1}$ | 7.69 10$^{-5}$ | 7.8 10$^{-5}$ | 7.70 10$^{-5}$ | 7.66 10$^{-5}$ | 7.65 10$^{-5}$ |
| $P_{PRZ}$ | MPa | 15.5 | 15.5 | 15.1 | 15.4 | 15.4 |
| Initial core power/$V_{PS}$ | kW/m$^3$ | 1123 | 1139* | 1453 | 1158 | 1165 |
| Break position | – | CL,<br>down stream<br>MCP | CL,<br>down stream<br>MCP | CL,<br>down stream<br>MCP | CL,<br>down stream<br>MCP | CL,<br>down stream<br>MCP |
| $P_{SIT}$ or $P_{accumulator}$ | MPa | 3.97 | 4 | 4.18 | 4.2 | 4.2 |
| $M_{SIT}/V_{PS}$<br><br>Injection point | kg/m$^3$ | 394.5**<br>CL | 124**<br>124**<br>DC | 117.6<br>119.4<br>CL | 113.1<br>113.6<br>CL | 606.7<br><br>CL |
| $M_{SIT}/V_{PS}$<br>(actually injected) | kg/m$^3$ | 232<br>CL | 118<br>118<br>DC | 192.05# | 202.17# | 197# |
| $A_{SG\ total}$/Core power | m$^2$/kW | 0.0496 | 0.066* | 0.046 | 0.048 | 0.056 |
| $P_{SS}$ | MPa | IL 6.94<br>BL 6.91 | 6.9 | 6.94 | 6.80<br>6.84<br>6.84 | 7.0 |
| $M_{SS}/M_{PS}$ initial | – | 1.42 | 1.33*** | – | 1.18 | 1.03 |
| $T_{rod}$ for LPIS actuation | K or MPa | 769 | 773 | 2.5 MPa | Not actuated | Not actuated |
| $G_{LPIS}/V_{PS}$ | kg/m$^3$s | 0.741 | 0.247<br>0.247<br>0.247 | 0.340<br>0<br>0.340 | – | – |
| Integral of core power<br>(20000 s)/$V_{PS}$ | MJ/m$^3$ | 544 | 544* | 575 | 569 | 559 |

* Core power includes power on by-pass
** Total mass of water in the accumulator
*** $M_{SS}$ – secondary side mass with the SG water level of 2.47 m; $M_{PS}$ – calculated primary side mass (including water in the pressurizer) with the pressurizer water level of 5.29 m; # – Total value; $V_{PS}$ – primary side volume without the pressurizer

**Table 3. Derived boundary and initial condition in the CT SBLOCA test**

| Parameter | LOBI | PSB-VVER |
|---|---|---|
| Primary side | | |
| Core power, [kW] | 630 | 1130 |
| Core by-pass power, [kW] | – | 15 |
| Primary system pressure, [MPa] | 15.47 | 15.5 |
| Reactor dT, [°C] | 28 | 28 |
| Loop outlet temperature, [°C] | 288 | 282 |
| Loop inlet temperature, [°C] | 316 | 310 |
| Pressurizer collapsed water level, [m] | 5 | 4.867 |
| Secondary side | | |
| Pressure, [MPa] | intact loop 6.94<br>broken loop 6.91 | 6.9 |
| Collapsed level, [m] | intact loop 8.14<br>broken loop 8.48 | 2.47 |
| ECCS accumulators | | |
| Pressure, [MPa] | 3.97 | 4 |
| Water volume, [m$^3$] | 0.222 | 2 0.202* |
| Water level, [m] | | 5.29 |
| Gas volume, [m$^3$] | 0.058 | 0.105 |
| Low pressure injection system | | |
| Mass flow rate, [kg/s] | 0.41** | 0.248** |
| Break equipment | | |
| Diameter, [mm] | 7.36 | 10 |
| Break size, [%] | 6 | 4.5 |

* SIT connected to the DC
** per each loop

**Table 4. Comparison between measured and calculated boundary and initial conditions**

| Parameter | Measurement | Measured value | Calculation R5/M3.3 |
|---|---|---|---|
| Pressure in upper plenum, [MPa] | YC01P17 | 15.6 | 15.53 |
| Coolant temperature at UP outlet, [°C] | YA01T03<br>YA02T03<br>YA03T03<br>YA04T03 | 310<br>308<br>311<br>308 | 310<br>310<br>310<br>310 |
| Coolant temperature at DC inlet, [°C] | YA01T02<br>YA02T02<br>YA03T02<br>YA04T02 | 283<br>283<br>282<br>282 | 284<br>284<br>284<br>284 |
| Core power, [kW] | YC01N01 | 1129 | 1130 |
| Core by-pass power, [kW] | YC01N02 | 14.9 | 15 |
| Coolant level in PRZ, [m] | YP01L02 | 4.93 | 4.87 |
| Pressure, [MPa]<br>– SG 1<br>– SG 2<br>– SG 3<br>– SG 4 | <br>YB01P01<br>YB02P01<br>YB03P01<br>YB04P01 | <br>6.88<br>6.91<br>6.93<br>6.88 | <br>6.92<br>6.92<br>6.91<br>6.92 |
| Level, [m]<br>– SG 1<br>– SG 2<br>– SG 3<br>– SG 4 | <br>YB01L01<br>YB02L01<br>YB03L01<br>YB04L01 | <br>2.48<br>2.49<br>2.52<br>2.48 | <br>2.45<br>2.44<br>2.44<br>2.44 |
| Pressure, [MPa]<br>– HA 2<br>– HA 4 | <br>TH02P01<br>TH04P01 | <br>4.08<br>4.14 | <br>4.0<br>4.0 |
| Level, [m]<br>– HA 2<br>– HA 4 | <br>TH02L01<br>TH04L01 | <br>4.58*<br>4.60* | <br>5.29<br>5.29 |

* Levels from the transducer readings do not cover the lower part of accumulator vessels

cilities (without the pressurizer), all the conditions were derived: pressurizer level, accumulators level, LPIS flow rate, *etc*. In tab. 3, the boundary and initial conditions of PSB compared with the LOBI ones are reported, while tab. 4 [15] shows the boundary and initial conditions measured during the test.

Regarding the break size, it must be noted that the percentage of the break area in PSB was different than in LOBI due to the different diameter of the cold leg in PWR and VVER 1000 prototype reactors.

The test started by opening the break valve, with the opening time of 0.4 s. With the opening of the break, the pressurizer heaters were switched off.

When the primary pressure achieved 13 Mpa, the following actions were imposed:
- closure of the turbine shut valve at the end of the steam header. The steam generators remained aligned to the steam header,
- closure of steam generator feed water,
- trip of all MCPs; completely stopped in 4 s, and
- scram, the core power started to follow the given power/time curve.

At the primary side pressure of 4 MPa the accumulators started to inject water in the pressure vessel inlet chamber. Accumulators remained connected until the water level was 1.31 m (from the bottom) in order to avoid nitrogen penetration into the primary system. Only the pair of SIT connected to the downcomer was used because LOBI had only two hydroaccumulators.

The low pressure emergency core cooling system (ECCS) was activated at the heater rod temperature of 500 °C. The water was supplied in cold legs of loops no. 1, 3, and 4 with the mass flow rate of 0.248 kg/s in each line.

The steam generator BRU-A valves were opened at the secondary pressure of 7.4 MPa and closed at 7.2 MPa; those set points were derived from the plots showing the secondary pressure behavior in LOBI facility. During the whole transient, the steam generators remained aligned to a common steam header; therefore, the secondary pressures of the steam generators were expected to be equal. An insignificant uncertainty in measuring the pressures could lead to a BRU-A valve being open in anyone steam generator while others remained closed, due to decrease of the secondary pressure. To avoid different behavior in the SGs, all the BRU-A valves were assumed to be dependent on SG-1 pressure. The experiment was terminated when a steady core cooling conditions after the final core rewet had been achieved.

## Experimental data comparison

After the experimental data collected at the PSB facility became available at University of Pisa, a comparison between LOBI and PSB experimental data has been done. From this first analysis the high similarity between the considered rigs can be seen (figs. 2 to 4). Further more, the main primary side parameters (such as pressure, clad temperature, mass) show the same trend. This fact confirms the validity of the approach followed in solving the scaling problem. Moreover, the same thermalhydraulic phenomena have occurred during the test (*e. g.* three dry-outs, the first quenched by loop seal clearing, the second by accumulator intervention and the last by LPIS intervention) – this demonstrates the similarity in behavior of PWR and VVER 1000 in the case of a SBLOCA.

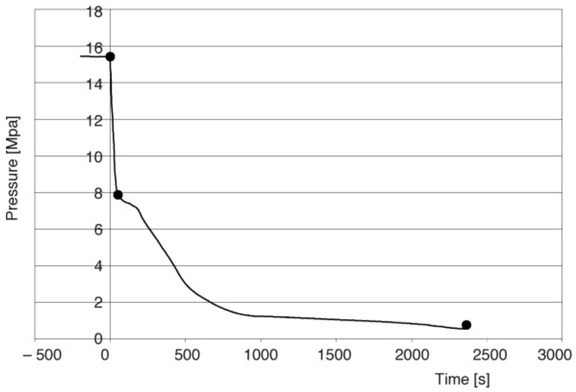


**Figure 2. Comparison between LOBI (full line) and PSB (dots) primary side pressure**

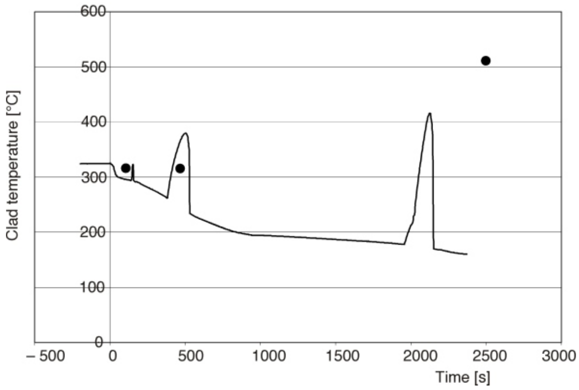


**Figure 3. Comparison between LOBI (full line) and PSB (dots) clad temperature – top level**

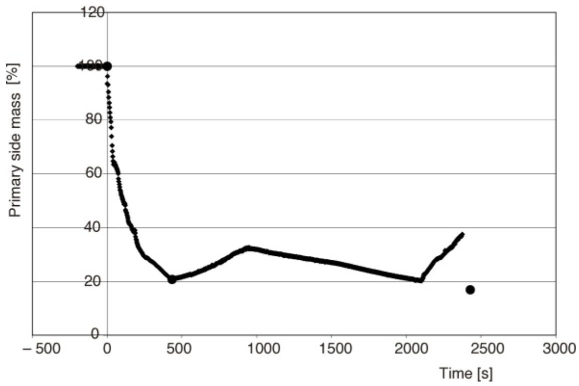


**Figure 4. Comparison between LOBI (full line) and PSB (dots) primary side mass**

From the phenomenological point of view, the accident can be subdivided into four main periods:

(a) subcooled blow down and first core dry-out/rewet (time from 0 to 97 s),
(b) saturated blow down and primary to secondary side pressure decoupling (from 97 s up to accumulators emptying),
(c) mass depletion in the primary loop (from accumulators emptying to the final core dry out), and
(d) intervention of the low pressure injection system that quenches the core.

*Phase (a).* Following the break, the primary system pressure is subject to an initial fast decrease up to the achievement of saturation conditions upstream the break. The abrupt initial pressure decrease leads to scram, main coolant pump trip and isolation of steam generators in the first 20 s of the transient. Pressurizer emptying occurs in about 10 s. During this phase, the stop of natural circulation occurs, essentially due to voiding and mass depletion in the upper zones of the loop. This causes a manometer type situation in the primary loop piping: the steam produced in the core partly flows directly to the break through the by-pass and partly pushes down the level in the core, to balance the liquid level present in loop seals. In this situation core dry out occurs at about 100 s. The rod temperature excursion ends when the loop seal clearing starts (at about 105 s in the loop number 4).

After the loop seal clearing occurs, the sufficient liquid mass is present in the core to cool the rods. Following the above events a large amount of steam is present upstream the break and an important break flow rate decrease takes place.

*Phase (b).* Continuous core boil off and primary to secondary side pressure decoupling characterize the first part of phase (b). The core boil off (produced steam flows almost entirely to the break) causes a second dry out at about 400 s at a pressure near, but higher than, the accumulator pressure (4.2 MPa). In this period the heat transfer from the secondary side to primary side is quite small compared with core power, because of the high void fraction in the SG tubes. The accumulator intervention at 406 s and 414 s causes the recovery of liquid level in the core and a second rewet that is completed at about 480 s. The isolation of accumulators occurs at about 1365 s and 1452 s respectively. During the period of accumulator injection, the primary system mass increases, because the liquid flow rate delivered by accumulators is larger than the break flow rate.

*Phase (c).* The stop of accumulator injection causes another mass depletion period, leading to the third dry out at about 2080 s into the transient, when the primary pressure is about 1 MPa. No other significant event occurs in this period, excluding the core level depression. When the rod surface temperature reaches 773 K, the low pressure injection system is actuated (2432 s) in the cold leg of the loop no. 1, 3, and 4.

*Phase (d)*. The LPIS flow rate (0.248 kg/s) is quite effective in causing the third core quench and in recovering the facility that occurs at about 2560 s. Core reflood occurs in this period.

The test was terminated at 2593 s, with primary pressure around 0.7 MPa.

## PSB counterpart test calculation results

Adopting the same nodalisation qualified at the steady state and at on-transient level by the 11% UP break test [17], the post test calculation of the SBLOCA has been performed. Little modifications have been done regarding the break area, its position, and the accumulator model. In fact, in this test the heat exchanged between the fluid and the accumulator structure has been taken into account due to the continuous discharge trend of the SITs.

In order to reach the steady state condition, the transient calculation has been preceded by running the code with the "TRANSNT" (transient) option for 200 s during which the plant has been maintained in the normal condition. The demonstration of the achievement of the right and stable steady state condition has been done and shown in tab. 4; the errors are acceptably small as requested from the nodalisation qualification procedure [16].

The post test calculation has been performed by adopting RELAP5/Mod3.3 code with the default model (Henry Fauske) for the break discharge. A comprehensive comparison between measured and calculated time trends or values shows that the code prediction is reasonably good [17, 18]. Nevertheless, some remarks should be made because of some discrepancies. For example, analyzing the rod temperature at the top level, it can be seen that the code does not predict the second dry out and that the break flow rate is not so precise. It is also true that not all the thermocouple installed at the core top measured three dry outs and that the entity of this phenomenon is quite low, about 50 K. It should be never forgotten that RELAP5 is a system code that gives a general trend and it is not able to predict or reproduce local effects. The flow rate through the break is measured with an error of about 20%.

To support the judgement of the calculation results, a qualitative and quantitative evaluations have been done. The first step has been completed following a systematic procedure, consisting of the identification of phenomena (CSNI list) and of relevant thermalhydraulic aspect (RTA). In both cases five levels of judgement have been introduced (E, R, M, U, and –), whose meaning is detailed in the Appendix 1 of [16]. The related results are reported in tab. 5.

A positive overall qualitative judgement is achieved if "U" is not present; in addition, the parameters characterizing the RTA (*i. e*. SVP – single valued parameter, TSE – parameter belonging to the time sequence of events, IPA – integral parameter and NDP – non dimensional parameter) give an idea of the amount of the discrepancy.

In the present case, the following conclusion could be reached:

(a) no "U" mark is present, and

(b) all RTA of the experiment are present in the calculated data.

The accuracy evaluation by adopting RTA and key phenomena supports the conclusion that the calculation is qualitatively correct.

**Table 5. Judgment of code calculation on the basis of RTA (part 1)**

| | | Unit | Experiment | Calculation R5/M3.3 | Judgment R5/M3.3 |
|---|---|---|---|---|---|
| RTA: Pressurizer empting | | | | | |
| TSE | Emptying time<br>Scram time | s<br>s | 10<br>57.6 | 10<br>53 | E<br>E |
| IPA | Integrated flow from SL (from 0 up to emptying) | kg | – | – | – |
| RTA: Steam generators secondary side behavior | | | | | |
| TSE | Main steam line valve closure<br>Difference between PS and SS pressure at 100 s | s<br>MPa | 17.5<br>0.33 | 10<br>0.43 | R<br>R |
| SVP | SG level | m | | | |
| | – at the end of subcooled blowdown | | 2.44; 2.44; 2.49; 2.38 | 2.39; 2.39; 2.38; 2.30 | E |
| | – when PS pressure equals SS pressure | | 2.43; 2.44; 2.49; 2.37 | 2.42; 2.43; 2.42; 2.33 | E |
| | – when accumulations start | | 2.40; 2.41; 2.47; 2.33 | 2.41; 2.41; 2.39; 2.31 | E |
| | – when LPIS starts | | 2.35; 2.26; 2.34; 2.30 | 2.28; 2.30; 2.30; 2.82 | E |

**Table 5. Judgment of code calculation on the basis of RTA (part 2)**

| | | Unit | Experiment | Calculation R5/M3.3 | Judgment R5/M3.3 |
|---|---|---|---|---|---|
| SVP | SG pressure<br>– at the end of subcooled blowdown<br>– when PS pressure equals SS pressure<br>– when accumulations start<br>– when LPIS starts | MPa | 7.31; 7.34; 7.36; 7.31<br>7.30; 7.34; 7.37; 7.30<br>6.79; 6.81; 6.83; 6.79<br>5.36; 5.39; 5.41; 5.37 | 7.25; 7.25; 7.25; 7.25<br>7.22; 7.22; 7.22; 7.22<br>6.87; 6.87; 6.87; 6.87<br>5.50; 5.50; 5.50; 5.50 | E<br>E<br>E<br>E |
| RTA: Subcooled blowdown | | | | | |
| TSE | Upper plenum in saturation conditions<br>Break two phase flow | s<br>s | 16<br>113 | 18<br>140 | E<br>R |
| IPA | Break flow up to 30 s | kg | 183.7 | 130 | M |
| RTA: First dryout occurrence | | | | | |
| TSE | Time of dry out<br>Range of dry out occurrence at various core levels<br>Peak cladding temperature | s<br>s<br>K | 97<br>97–102<br>589 | 127<br>127–131<br>619 | R<br>R<br>R |
| SVP | Average linear power<br>Maximum linear power<br>Core power/primary mass | kW/m<br>kW/m<br>kW/kg | 1.416<br>1.416<br>2.01 | 1.06<br>1.06<br>1.63 | M<br>M<br>M |
| IPA | Integral of dry out at 2/3 of core height | °C s | – | – | – |
| NDP | Primary mass/initial mass<br>Time of loop seal clearing | %<br>s | 47.6<br>109-105 loop 1&4 | 45<br>129-127 loop 3&4 | E<br>R/E |
| RTA: Rewet by loop seal clearing | | | | | |
| TSE | Range of rewet occurrence<br>Time when rewet is completed | s<br>s | 102–107<br>109 | 130–142<br>143 | R<br>R |
| TSE | PS pressure equal to SS pressure | s | 150 | 170 | R |
| SVP | Break flow at 200 s<br>Break flow at 1000 s | kg/s<br>kg/s | –<br>– | 1.04<br>0.13 | –<br>– |
| IPA | Integrated flow from 200 to 1000 s | kg | 236.52 | 196 | M |
| RTA: Mass distribution in primary side | | | | | |
| TSE | Time of minimum mass occurrence | s | 430<br>2430 | 429<br>2512 | E<br>E |
| SVP | Minimum primary side mass<br>Average linear power at minimum mass<br>Minimum mass/ITF volume | kg<br>kW/m<br>kg/m$^3$ | 171.3<br>140.6<br>0.304<br>– | 133<br>106.5<br>0.274<br>– | R<br>R<br>E<br>– |
| RTA: Second dryout occurrence | | | | | |
| TSE | Time of dry out<br>Range of dry occurrence at various core levels<br>Peak cladding temperature | s<br>s<br>K | 405<br>401-478<br>590 | –<br>–<br>– | M(1)<br>–<br>– |
| SVP | Average linear power<br>Core power/primary mass | kW/m<br>kW/kg | 0.425<br>1.41 | –<br>– | –<br>– |
| IPA | Integral of dry out at 2/3 of core height | | – | – | – |
| NDP | Primary mass/initial mass | % | 21.6 | – | – |
| RTA: Accumulators behavior | | | | | |
| TSE | Accumulators injection starts<br>Accumulators injection stops | s<br>s | 406-414<br>1365-1452 | 403<br>1702 | E<br>R |
| IPA | Total mass delivered accumulators | kg | – | – | – |

(1) Absence of second dry out is due to a low value of the dry out itself (about 50 K) and that is a local effect measured only by some thermocouple of the upper part of the core

The positive conclusion of the qualitative accuracy evaluation makes possible addressing the quantitative accuracy evaluation. With this aim, the methodology developed at University of Pisa, based on the use of the FFT, was adopted. The results are given in tab. 6, and the conclusion is the following: the achieved results are well below the acceptability threshold, both in relation to the overall accuracy (AA = 0.35 compared with the acceptability limit of 0.4) and the primary system pressure accuracy (AA = 0.06 compared with the acceptability limit of 0.1).

Definitely, the documented calculation is acceptable from the code assessment point of view; thus, the code is able to predict this kind of transient.

**Table 5. Judgment of code calculation on the basis of RTA (part 3)**

| | | Unit | Experiment | Calculation R5/M3.3 | Judgment R5/M3.3 |
|---|---|---|---|---|---|
| RTA: Accumulators behavior | | | | | |
| NDP | Minimum mass/initial mass<br>Primary mass/initial mass | %<br>% | 20.7<br>21.3 | 15.6<br>16.1 | M<br>M |
| RTA: Final dryout occurrence | | | | | |
| TSE | Time of dry out<br>Range of dry out occurrence at various core levels<br>Peak cladding temperature | s<br>s<br>K | 2077<br>2077-2313<br>783 | 2111<br>2120-2694<br>798 | E<br>R<br>E |
| SVP | Average linear power<br>Rate of rod temperature increase<br>Core power/primary mass | kW/m<br>K/s<br>kW/kg | 0.304<br>0.8<br>1.06 | 0.28<br>0.81<br>1.21 | R<br>E<br>R |
| IPA | Integral of dry out at 2/3 of core height | °C s | – | – | – |
| NDP | Primary mass.initial mass | % | 20.6 | 16.1 | R |
| RTA: LPIS intervention | | | | | |
| TSE | LPIS start<br>Range of rewet occurrence<br>Final rewetting | s<br>s<br>s | 2432<br>2482-2518<br>2559 | 2512<br>2535-2626<br>2695 | E<br>R<br>R |
| IPA | Integrated flow from start to end of rewet | kg | 96.7 | 135.8 | R |
| NDP | Primary mass/initial mass | % | 16.8 | 12.5 | R |

**Table 6. Results obtained by the application of the FFT method – SBLOCA**

| Parameter | AA | WF |
|---|---|---|
| PRZ pressure | 0.06 | 0.030 |
| Secondary side pressure – SG 3 | 0.08 | 0.052 |
| Secondary side pressure – SG 4 | 0.08 | 0.044 |
| Accumulator pressure | 0.09 | 0.026 |
| Core outlet fluid temperature | 0.45 | 0.086 |
| Upper head fluid temperature | 0.35 | 0.042 |
| Integral break flow rate | 0.20 | 0.114 |
| ECCS integral flow rate | 0.58 | 0.055 |
| Heater rod temperature (bottom level) | 0.08 | 0.039 |
| Heater rod temperature (middle level) | 0.56 | 0.035 |
| Heater rod temperature (top level) | 0.83 | 0.067 |
| Primary side total mass | 0.25 | 0.064 |
| DP core | 1.23 | 0.140 |
| Core power | 0.37 | 0.078 |
| DP loop seal ascending side (loop 4) | 0.55 | 0.115 |
| DP loop seal descending side (loop 4) | 0.28 | 0.040 |
| Total | 0.35 | 0.057 |

## CONCLUSIONS

The present study presents a contribution to the scaling analysis with special regard to the Russian pressurized water reactor type (VVER 1000). By the study of a counterpart test, strictly derived from the same experiment carried out in LOBI, the validity of the approach followed to solve the scaling problem in the PSB facility has been demonstrated. The selected test is a SBLOCA in which the break is located in the cold leg downstream the main circulation pump; the accumulators are available and only the intervention of low pressure injection system is foreseen, activated on a high temperature signal, regarding the active ECCS.

The comprehensive approach to the scaling issue is based upon a restricted number of key aspects:

(a) detailed scaling criteria have been derived: a few characteristic parameters are selected following a hierarchy based on their relevance in the considered transient, arriving at the definition of boundary and initial conditions for the selected test,

(b) comparison between experimental data of LOBI and PSB CTs available at University of Pisa has been conducted with the aims: (1) to verify the validity of the parameters selected, (2) to emphasize the same overall behavior in the two rigs, and (3) to include the CT performed in PSB facility that reproduces a VVER 1000 (unique CT carried out in this facility type) into the actual experimental database, and

(c) calculation of the SBLOCA carried out in PSB by the use of RELAP5/Mod3.3 code, comparison between measured and calculated curves, qualitative and quantitative evaluation of the results.

The scenario of the CT performed in PSB-VVER facility has been designed in collaboration with EREC, following the methodology described in the present study: the main parameter used to derive initial and boundary conditions has been the ratio between PSB and LOBI primary side volume (*Kv* scaling). The demonstration of the overall similarity of the two tests, even though the two facilities have different lay out, has been performed by a comparison between experimental data.

## LIST OF ABBREVIATION

AA – average amplitude
$A_R$ – break area
CSNI – Committee on the safety nuclear instalations
CT – counterpart test
DC – downcomer
DP – pressure drop
ECCS – emergency core cooling system
EREC – Electrogorsk research engineering center
FFT – fast Fourier transform
– hydro-accumulator
HPIS – high pressure injection system
IPA – integral parameter
ITF – integral test facility
*Kv* – volume scaling factor
LOBI – loop for off-normal behavior investigation
LPIS – low pressure injection system
MCP – main coolant pump
NDP – non dimensional parameter
NPP – nuclear power plant
PRZ – pressurizer
PSB – Russian large scale integral test facility
PWR – pressurized water reactor
RTA – relevant thermalhydraulic aspect
SBLOCA – small break loss of coolant accident
SETF – separate effect test facility
SG – steam generator
SIT – safety injection tank
SVP – single valued parameter
TACIS – technical assistance community independent state
TSE – time sequence of events
UP – upper plenum
$V_{PS}$ – primary side volume
VVER – Russian type of pressurized water reactor
WF – weighted frequency

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## АНАЛИЗА САГЛАСНИХ ТЕСТ ПОДАТАКА ИЗРАЧУНАТИХ И МЕРЕНИХ У СИМУЛАТОРИМА PWR И VVER 1000 РЕАКТОРА

У раду је приказан преглед "стратегије скалирања", посебно улоге усаглашене тест методологије. Да би се указало на појаве значајне за скалирање, изложена су нова проучавања која се баве анализама скалирања у лаководном реактору са посебним освртом на руски реактор VVER 1000. Усвојени приступ скалирању заснива се на одабиру неколико карактеристичних параметара на основу њиховог значаја за понашање транзијента. Коришћен је рачунарски програм RELAP5/Mod.3.3, и квалитативним и квантитативним проценама показана је његова тачност.

Упоређивањем експерименталних података утврђено је да се испитивана постројења слично понашају у погледу временских токова, и да се могу квалитативно предвидети исте термохидрауличке појаве. Главни резултат је да основни параметри PSB и LOBI постројења показују сличне трендове. Ова чињеница потврђује ваљаност усвојеног приступа скалирања и показује да је понашање PWR и VVER реакторских типова врло слично. Упркос чињеници да су два постројења различито пројектована, током усаглашеног тестирања нису уочене нове појаве, док су раније познате чињенице програмом исправно предвиђене. Могућности и тачност програма су независни од размере. Обе ове особине су нужне да би се у циљу предвиђања понашања нуклеарне електране допустили прорачуни у правој размери.

*Кључне речи: сигурност нуклеарног реактора, скејлинг анализа, VVER реактор, RELAP5*