SCALING ISSUES FOR THE EXPERIMENTAL CHARACTERIZATION OF REACTOR COOLANT SYSTEM IN INTEGRAL TEST FACILITIES AND ROLE OF SYSTEM CODE AS EXTRAPOLATION TOOL

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ABSTRACT

The phenomenological analyses and thermal hydraulic characterization of a nuclear reactor are the basis for its design and safety evaluation. In light of the impossibility and huge cost of performing meaningful experiments at full scale, scaled down experimental tests - Integral Effect Test (IET) and Separate Effect Test (SET) - are more feasible in developing "assessment database". The data are useful in characterizing the prototype design and in the validation of computational tools for safety analysis.

The analyses of system behaviors including component interactions in the Reactor Coolant System (RCS), the Containment System (PCV) and the RCS/PCV coupled system have been extensively investigated using IETs in the past decades. Though several scaling methods, e.g. Linear, Power/Volume, Three level scaling, H2TS..., have been developed and applied in the IET and SET design, a direct extrapolation of the data to the prototype, i.e. the scalability, is in general not possible due to unavoidable scaling distortions. The scaling distortions are related to many factors, mainly the complex geometry, multiple component interactions and two phase thermal hydraulic phenomena in steady state and transient condition of a nuclear reactor. The complex nature of scaling a nuclear reactor requires a large number of scaling parameters to be simultaneously fulfilled. In addition, physical construction and funding constraints demand that a scaling compromise is inevitable. Therefore a scaling approach, e.g. time preserved/not preserved, full height/reduced height, full pressure/reduced pressure, full power/reduced power…, has to be adopted in accordance with the objective of the IET or SET. Together with the scaling analysis, Best Estimate (BE) thermal hydraulic system code has been used for supporting experiment activity (design facilities, interpretation of results, etc) and for extrapolating results to full scale prototype conditions. Since the closure laws in the system code are mainly based on scaled test data, the extrapolation of code results remains a challenging and open issue.

Starting from a brief analysis of the main characteristics of IETs and SETFs, the main objective of this paper is to analyze some IET scaling approaches used to the simulation of RCS responses which characterize the main scaling limits. The scaling approaches and their constraints in ROSA-III, FIST and PIPER-ONE facility will be used to analyze their impact to the experimental prediction in Small Break LOCA counterpart tests. The liquid level behavior in the core and the core cladding temperature analysis are discussed used as judging criteria for the facilities scaling-up limits.

1. KEYWORDS

Scaling, integral test facility, nuclear safety analysis, reactor coolant system phenomena, containment phenomena, facility scaling-up limits, uncertainty

1. INTRODUCTION

The phenomenological analyses and thermal hydraulic characterization of a nuclear reactor are the basis for its design and safety evaluation. In light of the impossibility and huge cost of performing meaningful experiments at full scale, scaled down experimental tests [1-8] are more feasible in developing an “assessment database”. The database is useful in characterizing the prototype design and in the validation of computational tools for safety analysis [9-15], for supporting experimental test activity (design experimental facilities, interpretation of experimental results, etc) and for extrapolating experimental results to full scale prototype. In order to reproduce the behavior of a prototype reactor in a scaled-down model, it is necessary to thermal hydraulically characterize both the local and the integral phenomena. The test facility geometry and the initial and boundary conditions of the experiments should be correctly derived according to the scaling laws to avoid scaling distortions that could compromise the target phenomena identified through a Phenomena Identification and Ranking Table (PIRT) [16,17]. The Separate Effect Tests (SET) are used to characterize single phenomenon or combined phenomena bringing out the localized and isolated behavior in a system component (e.g., downcomer, pressurizer, hot and cold leg). The purpose is to reproduce localized prototype behaviors with minimum scaling distortions. Two kinds of SET could be identified: the Basic-SET where the single phenomenon/process could be characterized; Component-SET where the thermal hydraulic behavior of a reactor-component and the related local phenomena/processes could be characterized [8-11,13]. The Integral Effect Tests (IET) are used to characterize the integrated system responses of a reactor region (e.g., RCS and PCV), including the interactions between different phenomena and components [8,9,12,13,15]. The data obtained from both SETs and IETs can be used in model development/improvement and code assessment.

The analyses of system behaviors and component interactions in the RCS [8,9,12,13], the PCV [15] and the RCS/PCV coupled system [14] have been extensively investigated using IETs in the past decades. Though several scaling methods, e.g. Linear, Power/Volume, Three level scaling, H2TS..., have been developed and applied in the IET and SET design, a direct extrapolation of the data to the prototype, i.e. the scalability, is in general not possible due to unavoidable scaling distortions. The scaling distortion is related to many factors, mainly the complex geometry, multiple component interactions and two phase thermal hydraulic phenomena in steady state and transient condition of a nuclear reactor. The complex nature of scaling a nuclear reactor requires a large number of scaling parameters to be simultaneously fulfilled. In addition, physical construction and funding constraints demand that a scaling compromise is inevitable. Therefore a scaling approach, e.g. time preserved/not preserved, full height/reduced height, full pressure/reduced pressure, full power/reduced…, has to be adopted in accordance with the objective of the IET or SET.

Starting from a brief analysis of the main characteristics of IETs and SETFs, the main objective of this paper is to analyze some IET scaling approaches used in RCS which characterize the main scaling limits. The scaling approaches and their constraints in ROSA-III, FIST and PIPER-ONE facility will be used to analyze their impact to the experimental prediction in Small Break LOCA counterpart tests. The liquid level behavior in the core and the core cladding temperature analysis are discussed as judging criteria for the facilities scaling-up limits. A detailed analysis of the key elements necessary for developing and using the “assessment database”, and the related interactions, are here synthetized and represented in Fig. 1.

2. USE OF THE ASSESSMENT DATABASE IN NUCLEAR REACTOR TECHNOLOGY

2.1. Scaling Methods in Nuclear Technology and Unavoidable Facility Distortions

When a scaled-down test facility is designed, a scaling analysis [1-7,16,17] is necessary to assure that the experimental data obtained are representative of the physical behavior of the prototype (Fig. 1 block 1). In a nuclear reactor we are in presence of a complex geometry with two-phase flow where more than one phenomenon takes place [17]. This determines that a large number of scaling parameters should be preserved at the same time. It's then difficult to have a complete and consistent set of scaling criteria. Therefore it is difficult to apply the dimensional analyses and a simplified approach is necessary. In order to assure that the *"measured transient"* be representative of the full scale prototype expected transient evolution, the main targets of this kind of simplified scaling analysis should be: (A) Full scale dominant/relevant phenomena and their interaction should be characterized in the facility; (B) Significant phenomena have to play the same role in the scaled-down test facility and in the prototype (phenomena weight is correct). Therefore the scaling analyses of a complex system (Fig. 1 block 2), with multiple component interactions and two phase thermal hydraulic phenomena, have to determine: (i) Which are the dominant and relevant phenomena that should be investigated [PIRT]; (ii) The independent dimensionless groups that should be preserved. This simplified approach permits that the scaled-down facility "measured transient" should be characterized by the same dominant and relevant phenomena of the full scale prototype "real transient"; therefore the main distortions should be related only to the not relevant phenomena, as long as the initial scaling analysis correctly provided prototype conditions and results [1-17].

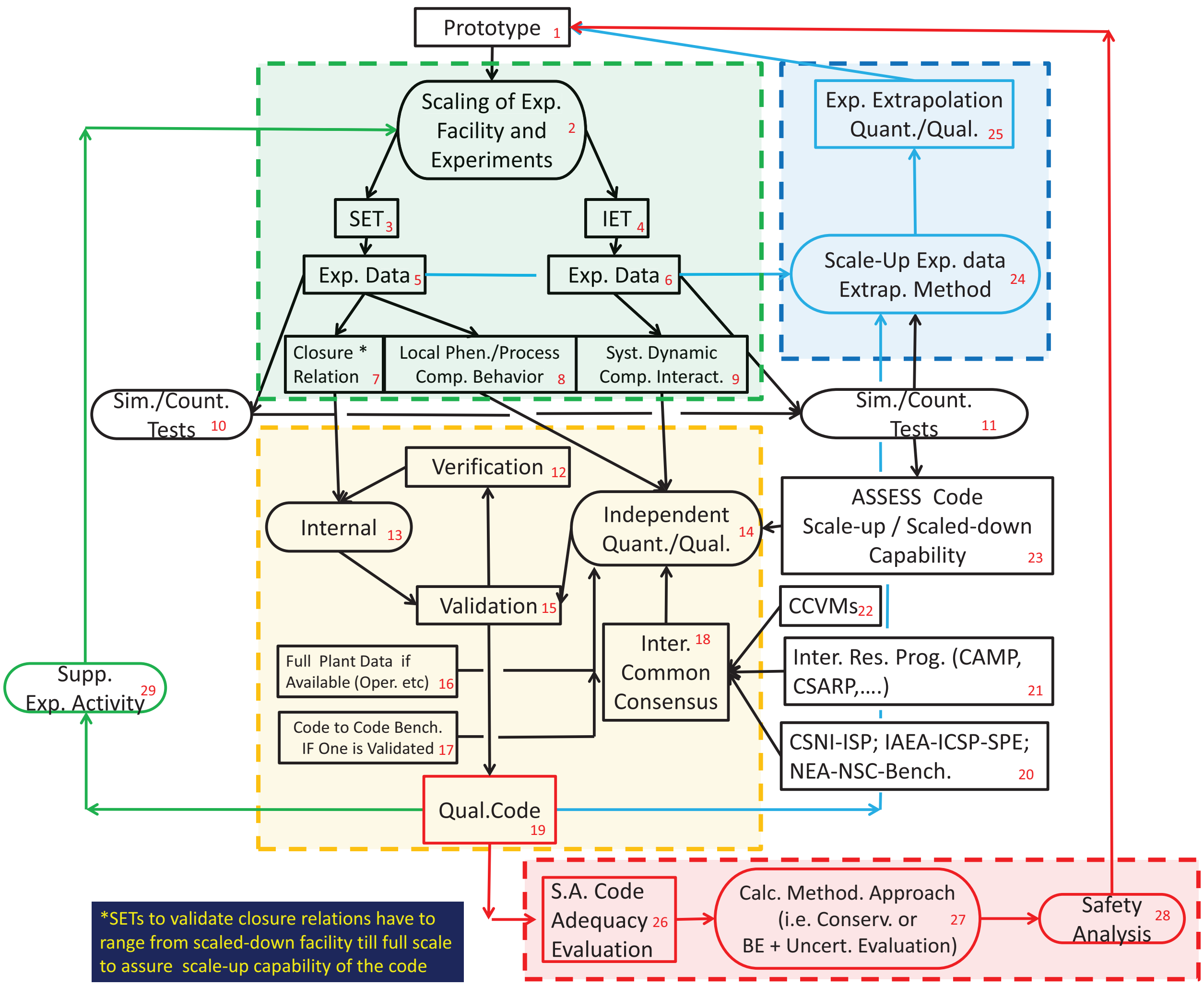


Figure 1. Key elements necessary for developing and using the "assessment database".

In general a scaling distortion is a *"deviation"* in the scaled-down facility design that has an effect the *"partial or total suppression of a phenomenon"* [5]; this causes discrepancies in experimental observation from prototype physical behavior. The distortions related to SET facility (Fig. 1 block 3) are mainly due to the imposed boundary conditions, initial conditions deviation, local geometrical distortions and fluid scaling if non prototypical operational fluid is used. In relation to the IET (Fig. 1 block 4) the main distortions are mainly due to the differences in the employed scaling approaches (i.e. reduced volume; reduced/not reduced height; reduced/not reduced pressure, lumped/non lumped loop, etc). Considering FIST/ROSA-III/PIPER-ONE facilities, used for the BWR counterpart test activity presented in this paper, the main facility scaling properties are reported in Table I. Of particular interest are the scaling methods used, the volumetric and height scaling of the facility, the fluid used and the facility operation conditions. All the three facilities have been designed with Power/Volume scaling method to have a real time transient simulation (Time preserved approach). The volumetric scaling choice of FIST (1:624) and PIPER-ONE (1:2200) determine an increase of the (surface area)/(hydraulic volume) ratio causing distortion in the "stored metal structures". Since this can affect total steam generation rate in the facility as deviation, a larger ADS flow area is considered in these two facilities to compensate these discrepancies. In ROSA-III facility the height reduced approach used can causes distortions in the void fraction distribution. In ROSA-III, FIST and PIPER-ONE the operation conditions and the fluid are prototypical.

In general, facility unavoidable distortions determine the *"facilities scaling-up limits"* precluding the possibility of a direct application of the experimental results to the full scale prototype – experimental data scalability –. Therefore, though an experimental test facility and an experiment are designed with rationale scaling criteria, obtained results should be limited within the experiments conditions and test scale. An extrapolation procedure, *"Scale-up Experimental Data Extrapolation Methodology"*, Fig. 1 blue blocks, is always necessary for scaling-up the experimental data. In general, in the nuclear sector, computational tools are the key component of the extrapolation methodology that could be applied as long as models and criterion used in the computational tools (computer codes) are validated up to the prototypical conditions.

2.2. Code Validation Process

The computer code, in general, has to reproduce the full scale prototype physical behavior in steady and transient condition [18-20]. This is characterized by overall system dynamic, system component interactions (Fig. 1 block 9) and local component phenomena/processes (Fig. 1 block 8). Since it is difficult to perform thermal hydraulic test at full scale, the code is usually validated against scaled-down test data, Fig. 1 orange blocks. Within this regard, the code has to be able to reproduce the qualitatively and quantitatively physical behavior of experimental test facilities of different scales till full scale prototype condition. It is to underline that plant data (operational transient, start-up test, etc), not affected by scaling issue, are used for the assessment of the code (Fig. 1 block 16) but only a few parameters are measured in comparison with an experimental test facility that is more suitable for the code assessment [12]. A "Code to Code Benchmark" (Fig. 1 block 17) is possible only if one of the codes is already validated for the target phenomena/processes [21-22]. A best estimate thermal hydraulic system code, for example, is a computational tool based on the "Two Phase Non Equilibrium Model". This model is based on 2 mass conservation equations, 2 momentum conservation equations and 2 energy conservation equations; each conservation equation is applied for the liquid and vapor phases – 6 equation model with α, P, v_l, v_v, T_l, T_v, calculated parameters. Constitutive relations are "flow regime dependent" and are used to characterize the wall friction and heat transfer, interfacial mass/momentum/energy transport [23]. It is to underline that since the constitutive equations are "flow regime dependent", in order to develop and validate these models, SETs in different and several scales are necessary. Since the results of the experimental data are related to the facility scales, constitutive equation has to provide a realistic description of the phenomena/processes with the correct scale feedback. Therefore SETs to validate such constitutive equations (Fig. 1 block 7) have to range from scale-down facility till full scale to assure the scale-up capability of the computer code into which the constitutive equations are utilized.

To assess the quality of the code prediction a “Code Validation Process” (code validation is also called code qualification or code assessment [22]) has to be fulfilled (Fig. 1 block 15) [18-20]. As underlined in Fig. 1, the main parts of the “Code Validation Process” are the “Code Internal Development” and “Code Independent Qualification”. The “Code Verification” (Fig. 1 block 12) and the “Internal Code Validation” (Fig. 1 block 13) constitute the “Code Internal Development”; main target is to qualify the model implemented in the code and the global code architecture (i.e. verification of code design and source code; individuation of errors and related corrective actions etc) [22]. The main target of the “Code Independent Qualification” (Fig. 1 block 14) is to evaluate the code accuracy - accuracy is connected with the error in the comparison between measured and calculated trends - . This evaluation is done by an “independent user” by comparing the “calculated transient” against the “measured transient” developed in a scaled-down test facility. An “Independent Qualitative and Quantitative assessment” could be identified [18-20]. In order to evaluate the qualitative and quantitative code accuracy, the transient’s ranges that should be covered by the code and the related important target phenomena/processes have to be identified. Then, to evaluate the code accuracy qualitatively, the “Test Case” has to be investigated by [18]:

1) Identifying the “Relevant Thermal Hydraulic Phenomena”. Within this regard the “International Recognized Code Validation Matrix” has a key role. In fact, through the experience of an “International Recognized Group of Experts” the relevant phenomena suitable for code assessment, the phenomenon occurrence vs. test type (Sub Matrix 1- SM1), the suitability of the facility for code assessment vs. phenomenon (SM2), and the correlation of the test facility and test type (SM3) have been considered. In this paper a “SBLOCA (break area=2.6% A max) in one recirculation loop” is one of the tests investigated. The main phenomena investigated in PIPER-ONE facility during this test are: core dry-out, CCFL at Upper Core Support Plate (UCSP) and channel inlet orifices, heat transfer in partially uncovered core, fuel rod quenching, ECC mixing [12].
2) Identifying the “Phenomenological Windows (*PhW*)”, characterizing the selected scenario: it consists in “time spans” in which a unique relevant physical process mostly occurs, and a limited set parameters control the scenario; the dominant phenomena consequent to the physical processes characterize each PhW. For example in the SBLOCA (break area=2.6% A max) test three PhWs could be identified: the first PhW is characterized by a constant pressure due to the regulation; the second PhW of the test starts when the ADS actuation takes place; the third PhW of the test starts with the ECCS injection to refill the core.
3) Identifying the “Relevant Thermal Hydraulic Aspects (*RTA*)” inside each PhW. These are the events or phenomena consequent to the physical process. These are peculiar of the transient investigated. The selection of *“RTA characterizing parameter”* is necessary to have quantitative information on it: “Single-Valued Parameters” (SVP), “Non Dimensional Parameter” (NDP), etc. For example in the SBLOCA (break area=2.6% A max) some of the “RTA characterizing parameter” selected are the Peak Cladding Temperature (PCT) and its occurrence, time of ADS actuation, time of top of bundle uncover , etc. Table II summarizes the main sequence of event of the transient.
4) Qualitative analyses of obtained results by evaluating and ranking the comparison between measured and calculated trend (Continuous Valued Parameter-CVP). For example for the SBLOCA (break area=2.6% A max) the steam dome pressure time evolution, downcomer level time evolution, ECCS mass flow rate time evolution, residual mass in the loop time evolution, rod surface temperature time evolution have been selected.

Qualitative analysis is based on five subjective judgment mark (Excellent, Reasonable, Minimal, Unqualified, not applicable), that are applied both to the matrix of phenomena and to the list of RTA. It is mainly based from visual observation of the experimental and calculated trends. The evaluation of the qualitative code accuracy will be based on a comprehensive comparison between experimental and calculated data including the following steps: (A) Comparison between experimental and calculated trend; (B) Comparing quantities characterizing the calculated sequence of event; (C) Qualitative evaluation of the calculation accuracy on the basis of the phenomena included in the CSNI matrix; (D) Qualitative evaluation of the calculation accuracy on the basis of RTA (this could be used in some methodology also for code uncertainty derivation). The positive conclusion of the “Qualitative Accuracy Assessment” permits the analyses of the “Quantitative Accuracy Assessment” as a number- Quantitative Judgement of the Code Accuracy -. The Fast Fourier Based Transform Method is an example.

Another component of the “Code Independent Qualification” is the “Assessment of the Scaling-up Code Capability”, Fig. 1 block 23. Since the main target of a code is to predict the transient behavior of a full scale reactor, one way to try to assess the scaling-up capability of the code is to analyze different tests at different scales with the same or similar initial and boundary conditions. These tests are called “Similar/Counterpart test” [24] (Fig. 1 blocks 10, 11). The general conditions for Similar/Counterpart tests are reported in [12]. In this framework the role of counterpart tests is to provide data to assess the capability of the code to predict the same phenomena at different scales. For example in the SBLOCA (break area=2.6% A max) three different facilities with different volumetric scales, designed with the same Power/Volume method but different scaling approaches (reduced height approach for ROSA-III) are used to characterize, at different scales, phenomena as: channel and bypass axial flow and void distribution; core heat transfer including departure from nucleate boiling (DNB), dryout, RNB, surface to surface radiation; etc.

The “International Common Consensus”, Fig. 1 block 18, is one of the most important parts of the “Code Independent Assessment”; it consists in the international cooperation platform of research activities where exchange of opinions, methods, experimental/calculated data and idea takes place. Examples are the International Standard Problem (ISP) of the OECD–NEA [25] and the International Collaborative Standard Problem (ICSP) of the IAEA [26], Fig. 1 block 20. The ISP are important part of the qualitative code assessment because permit a better understanding of postulated events, to compare and evaluate the capability of the code, to suggest improvements to the code developers, to improve the ability of code users and to address the scaling effect [20]; the same considerations are valid for the IAEA-ICSP. Other examples of the “International Common Consensus” are the USNRC research programs (Fig. 1 block 21) as Code Applications and Maintenance Program (CAMP) and Cooperative Severe Accident Research Program (CSARP) [27]. Another part of the “International Common Consensus” is the international recognize “Code Validation Matrixs” (Fig. 1 block 22) to be used for the validation of codes [10-15].

2.3. Full Scale Plant Code Prediction

At the end of the “Validation Process” a “Qualified Frozen Code” (Fig. 1 block 19) is obtained and distributed to the “International User-Code Community” ready for full scale plant application. The limits of the code and its capability and the validation range have to be well know/acceptable and documented. This code is ready to be used for supporting experimental test facility program (scaled–down capability of the code [6]), Fig. 1 block 29, for extrapolating experimental results to full scale reactor, Fig. 1 blue blocks, and for full scale plant application, Fig. 1 red blocks. For example in relation to the THYDE-B1 code, used also in the analyses presented in the next section, it is well known and documented that it is one-dimensional lumped—parameter code designed to simulate, with a coarse nodalization, the pressure and core mixture level behavior in a BWR during a small/intermediate- break LOCA where fluid motions are essentially gravity-controlled. The code has no capability for the simulation of the CCFL phenomena and the use of the Wilson correlation, derived from experiments in circular tube and used for predicting region different from the core, could cause some discrepancies between calculated and experimental data. For example the prediction of the measured mixture level of the DC after flashing initiation due at the ADS actuation is good for FIST data, characterized by an external/circular DC, but it is poor for ROSA-III data characterized by an internal/annular DC. Therefore the full scale behavior of the mixture level in the DC cannot be predicted by the code.

As underlined before facility unavoidable distortions determine the *“Facilities Scaling-up Limits”* precluding the possibility of a direct application of the experimental results to the full scale prototype.

Experimental results obtained from counterpart tests activity should be limited, also, within the maximum condition range. Since results obtained by SET or IET are limited within the experimental conditions and test scale, the code is validated and can be used only inside the experimental range investigated, *"Code Scaling-up Issue"*. Therefore, since there is a quite limited number of experimental data at full scale, the application of computational tools for the prediction of full scale prototypical behavior and the related methodology is still an open issue. The application of a code, outside of its validation range (full scale plant analysis or extrapolation tool) requires caution and should be inserted in a *"Well-defined Extrapolation Methodology"*, Fig. 1 blue blocks. For example a methodology to attempt to scale the accuracy of the RELAP5/Mod2 code for the prediction of BWR full scale prototype against the ROSA-III/FIST/PIPER-ONE counterpart test data is in briefly described in the next section. Another methodology is applied to attempt to predict the BWR full scale behavior by using the THYDE-B1 code against ROSA-III and FIST counterpart test data. These are two examples of "Scaling-up Experimental Data Extrapolation Methodology" to permit, by "Assessing the Scaling-up capability of the computer code" against counterpart test data, the "Quantitative and Qualitative Extrapolation of the Experimental data" to characterize the full scale prototype behavior for the transient condition experimentally investigated. The same considerations are valid to extrapolate single facility experimental data.

In this framework has to be considered the *"Deterministic Safety Analyses"* for *"Safety Review"* purpose, red blocks in Fig.1. The application of the BE thermal hydraulic system code for the "Deterministic Safety Assessment" is a direct link between Scaling and "Safety Review". The application of a "Qualified Code" for "Safety review" requires that a "Safety Analyses Code Adequacy Evaluation [21]" (Fig. 1 block 26) should be satisfied for demonstrating its qualification level. Within this regard, although a code fulfills the qualitative/quantitative accuracy evaluation (Qualified Code) and the "Safety Analyses Code Adequacy Evaluation", its plant results are still characterized by uncertainty (uncertainty is used as a measure of the error made with a code in prediction plant behavior); one of the uncertainty source is related to the "scaling issue" [19]. Therefore, going beyond the "Well-defined Extrapolation Methodology", the use of a code "Qualified and Adequate for Safety Evaluation" for safety review has to be coupled with a "Well-Defined Calculation Approach/Methodology" (conservativeness or BE + Uncertainty evaluation), Fig. 1 block 27. In general the application of a thermal hydraulic system code implies the choice of the following calculation methodologies/approaches [19,22]: (1) Conservative computer code, conservative assumption for availability of the system, conservative boundary and initial condition (BIC); (2) Best Estimate (BE) computer code, conservative assumption for availability of the system, conservative BIC; (3) BE computer code, conservative assumption for availability of the system, realistic BIC with uncertainty; (4) BE computer code, PSA based assumption for availability of the system, realistic BIC with uncertainty. For example, for an adequate uncertainty evaluation methods, a "Qualified Code" is adequate for safety review if fulfill the "Bottom-up Adequacy" (Pedigree, Applicability, Fidelity, Scalability) and the "Top-down Adequacy" (Numeric, Fidelity, Applicability) as reported in [21]. It is to underline that "Safety Analyses Code Adequacy Evaluation" and "Well-Defined Calculation Approach/Methodology" are very close related and connected.

3. EXPERIMENTAL DATA EXTRAPOLATION PROBLEMS, COUNTERPART TEST AND ROLE OF SYSTEM CODE

In relation to the previously mentioned BWR counterpart test activities, a 2.8 % recirculation pump suction line break LOCA counterpart test has been performed in FIST and ROSA-III, the results and the code application has been detailed reported in [28]. A "SBLOCA (break area=2.6% A max) in one recirculation loop" counterpart test, assuming the unavailability of the emergency high-pressure core spray system, has been performed in FIST, ROSA-III and PIPER-ONE, the results and the code application has been detailed reported in [29,30,31]. The main scaling properties of the FIST/ ROSA-III/ PIPER-ONE facilities are presented in Table I. Table II shows the sequence of events of the FIST/ROSA-III/PIPER-ONE counterpart test.

Table I. Main scaling properties of FIST, ROSA-III, PIPER-ONE [9,29,32].

Quantity	FIST	ROSA-III	PIPER-ONE	BWR-6
Reference Reactor	GE-BWR/6	GE-BWR/6	GE-BWR-4/6	-
Scaling Method	Power/Volume	Power/Volume	Power/Volume	-
Volumetric Scaling	1:624	1:424	1:2200	1:1
Height Scaling	1:1	1:2	1:1	1:1
Pressure Scaling	Full Pressure	Full Pressure	Full Pressure	Full Pressure
Power Scaling	Full Power	Decay Power*	Decay Power	Full Power
Core Heating Method	Non Nuclear	Non Nuclear	Non Nuclear	Nuclear
Recirculation Loop Scaling	2	2	-	2
Jet Pump Scaling	2	4	1	24
Fluid Scaling	Steam-Water	Steam-Water	Steam-Water	Steam-Water
Primary Volume (m^3)	0.67	1.42	0.19	620
Top of RPV (m)	19.42	6.04	13.78	21.30
Core Heated Length (m)	3.81	1.88	3.71	3.81
Bundle Array Type	8×8	8×8	4×4	8×8
Number of Bundles	1	4	1	624
Maximum Power (MW)	5.05	4.46	0.25	3150
Core Rod Number	64	284	16	52576
Ext. Rod Diameter (mm)	12.3	12.3	12.3	12.3
Pitch (mm)	16.2	16.2	16.2	16.2
Core Heating Method	Skin	Indirect	Indirect	-
Local Peaking Factor	1.04	1.1	1	1.13
Radial Peaking Factor	1	1.4	1	1.4
Axial Peaking factor	1.4	1.4	1.26	1.4
DC Pos/Shape	Ext/cyl	Int/ann	Ext/cyl	Int/ann
DC Volume (m^3)	0.170	0.394	0.042	108.4
Scaling of DC volume (Y/Y_R)	1:638	1:275	1:2580	1:1

* Decay power was used up to 44% for power supply limitation.

In relation to the SBLOCA FIST/ROSA-III counterpart test analysis [28], it is to underline that no conflicting phenomena are observed in the two experiments. Some differences are observed in the timing of the phenomena. These are due to a) experiment specifications difference and b) facility scaling differences. In relation to the point a) in FIST the pressure control is set to maintain the RPV pressure at the initial pressure of 7.23 MPa, in ROSA-III the pressure control is set to prevent the pressure decrease below 6.7 MPa; this causes some discrepancies in the period between the break initiation and the MSIV closure. In relation to the point b):

- ADS actuation takes place 50s later in ROSA-III because to its larger initial downcomer water inventory and the low pressure set point;
- The depressurization after the ADS actuation is slower in ROSA-III due to the FIST oversized ADS flow rate to compensate the distortion due to the stored energy release from the structure. This should be coupled with the larger initial water mass in ROSA-III;
- The inside-shroud mixture level swell after the ADS actuation is smaller in ROSA-III facility. This is due to the larger flow area in ROSA-III facility that lets vapor generated in the mixture volume leave from the mixture level more rapidly that in the other facility. The slower depressurization after the ADS actuation in the ROSA-III can have an influence also.
- Mixture level swell after the flashing initiation in the FIST test is larger than that in the ROSA-III test. One probable reason is that the downcomer of the ROSA-III is shorter and wider than that of

- FIST and the mixture level swell is smaller. Another reason is that the downcomer of ROSA-III is an annulus like a BWR's downcomer, whereas the downcomer of the FIST is a pipe.
- PCT, that is an important judging criterion for safety margin discussion, takes place at the core mid plane and has the same magnitude in ROSA-III/FIST facilities.

A detail analysis of the phenomena taking place during the test is reported in the [28].

In relation to the simulation of the SBLOCA counterpart test (Area=2.6%Amax) with FIST/ROSA-III/PIPER-ONE facility, detailed analyses of the Initial and Boundary Condition are reported in [29]. As underlined before, three "Phenomenological Windows" could be identified. The *first phase* is characterized by a constant pressure due to the regulation of the steam line for FIST/ROSA-III and steam relief valve for PIPER-ONE. As underlined before in the ROSA-III facility the pressure set point is 6.7 MPa; in FIST the set point of the pressure system is slightly higher. In the PIPER-ONE the pressure regulation device is not available therefore the steam relief valve is used. The main phenomenological difference between the facilities "measured transient" is a first dry out in the PIPER-ONE facility during this first phase. This first dry out doesn't appear in the other two facilities. The reasons are a) the greater value of the overall energy input to the fluid in this period b) the faster occurrence of the mass depletion in the downcomer, resulting from an initial inventory that is lower than the ideally scaled value and from the higher flow rate between the downcomer and the lower plenum. The *second phase* of the test starts when the ADS actuation takes place. This determines a fast depressurization of the system and an increase of the rate of coolant loss. Core uncovers and rod heat up occurs during this period in all the tests. In PIPER-ONE the ADS actuation quenches the early dry out, but a second dry out soon occurs in the highest part of the core simulator. The second dry out is quenched during the depressurization in PIPER-ONE, while only ECCS injection is effective in the other two experiments. The *third phase* of the test starts with the ECCS injection to refill the core. In relation to the PIPER-ONE the heat release from the structures lead to a small increase in the system pressure at about 400s, which also affects the ECCS flow rate. The sub-cooled liquid coming into the core causes the rod surface temperature to decrease below the saturation temperature [29,30,31].

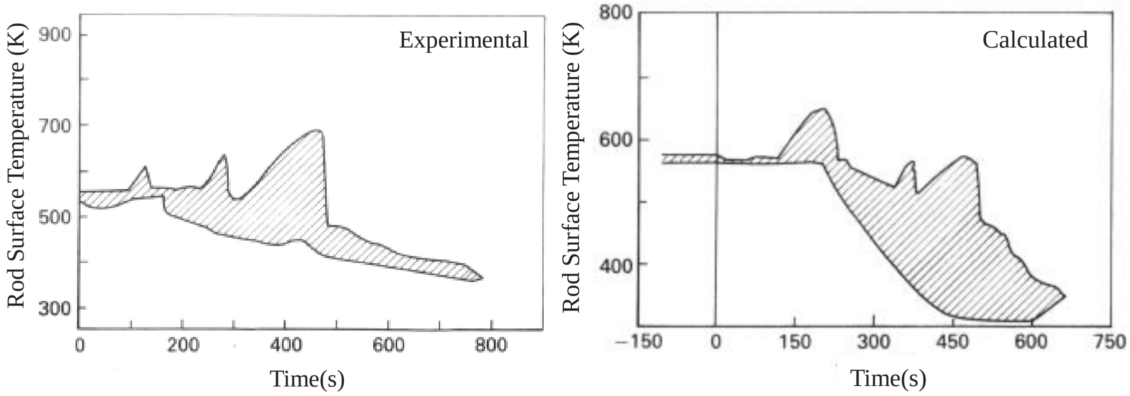
The judgment of the similarity of the experimental data, "Similarity Analyses", is one the necessary step to apply the experimental data for the assessment of code capability to predict similar phenomena at different scales. Within this regard it is possible to consider: (A) Key phenomena or Relevant Thermal Hydraulic Aspects in the different facilities (i.e. parameter characterizing the phenomena, etc…..); (B) Single-Valued Parameters (SVP) prediction in the different facilities (i.e. *SVP vs. Kv*, etc); (C) Continuous Value Parameter (CVP) prediction in the different facility (i.e. *CVP vs. Time*, experimental dispersion band,…). The envelope of a selected CVP for the different counterpart test facilities determines the "Experimental Dispersion Band". Figure 2(a) shows the experimental dispersion bands related to the "rod surface temperature at the 7 level (top elevation in the core heated length)" for the FIST/ROSA-III/PIPER-ONE facilities. Larger is the shaded area, bigger is the discrepancies among the counterpart test facility results. Other scaling consideration about the similarity of the results could be obtained by analyzing *"SVP vs. Kv"*. Figure 3 shows PCT and the lower plenum flashing (LPF) occurrence against the Kv of the facilities. Detailed analyses of the similarity of the experimental trends are reported in [29,30,31] and demonstrate that sufficient knowledge has been obtained in fixing the design scaling law (well posed scaling approaches, well defined and recognized scaling methods, facility biases identified/characterized/understood/if possible decrease their effect) of the IETs and the counterpart test criteria utilized to specify the boundary and initial condition of the experiments.

This "similarity analysis" could be considered also as an "attempt" to have a direct extrapolation of the experimental counterpart test results to the full scale prototype. As underlined before however, considering the unavoidable scaling distortions and experiment specifications differences, the experimental results obtained from counterpart test activity should be limited within the maximum condition range delimited by the bigger facility. Although some phenomenological consideration could be obtained from counterpart test activity (i.e. PCT takes place at the core mid plane and has the same magnitude in ROSA-III/FIST facilities) from the analyses of the experimental results of the ROSA-III/ FIST/ PIPER-ONE and from other counterpart test activity related to PWR [29,30,31], it is confirmed that the attempt to have a direct extrapolation (scaling-up) of the experimental data to the full scale prototype is not in general possible [29,30,31]. In fact, for example, single valued parameters and key phenomena or relevant thermal hydraulic aspects are characterized mainly by data randomly dispersed, for example in a hypothetical *SVP vs. Kv plot*. In order to scaling-up the "measured transient" to the "expected prototype transient" an extrapolation methodology is always necessary. This is valid for single facility experiment and for counterpart test experiments. As underlined before, the key component of this "extrapolation methodology" is the code.

Table II. Main sequence of events for FIST, ROSA-III, PIPER-ONE [29].

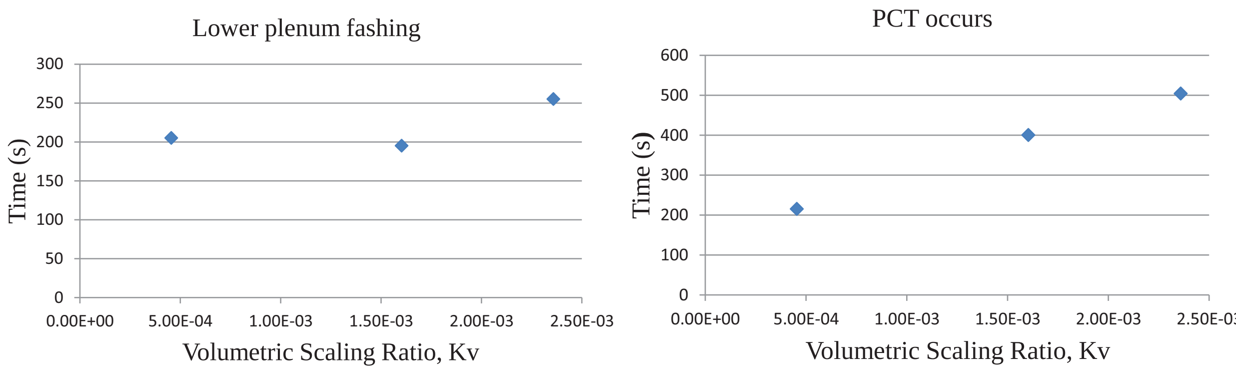
Event	PIPER-ONE (Kv= 4.55E-04)	FIST (Kv=1.60E-03)	ROSA-III (Kv=2.36E-03)
Break initiation	9	0	0
Pump trip	-	0	0
Feed water line trip	-	0	3
Power decay begins	50	0	10
MSIV trip	55	77	131
ADS actuation	182	195	249
Lower plenum flashing	205	195	255
Top of fuel bundle uncovers	110	237	329
Bundle dry out begins	117	250	330
LPCS flow begins	263	310	469
Bottom of core uncovers	NA	325	458
LPCI flow begins	295	335	502
Bundle refill begins	290	370	490
PCT occurs	215	400	504
Bundle quenched	230	420	527
Bundle refill completed	340	420	528
Lower plenum refilled	290	465	NA
End of test	500	508	788

Two examples, related to the BWR counterpart test activity, of code as extrapolation tool are briefly reported hereafter. In the first example [28], the THYDE-B1 code has been used against the FIST and ROSA-III counterpart test to assess its scaling-up capability to predicted BWR SBLOCA transient. The basic sequence of events and key phenomena in the ROSA-III/FIST counterpart test are similar, being individuated and understood the facility scaling distortion effect on the experimental results. After the validation of the code capability and limits to predict the facilities "measured transient" and the conclusion that code reproduces both test results quite well, the simulation of the BWR under the boundary conditions the same as those in the counterpart test scenario has been executed by the code. The analyses showed that the code has the capability to predict accurately the thermal hydraulic response during BWR SBLOCA that appeared in both experiments, though the models for the prediction of mixture level in annulus DC and the post-dryout heat transfer coefficient during LPCS spray actuation were then considered subject to improve. For example, the PCT predicted in the experimental tests is 710K in ROSA-III and 769K in FIST and occurred near the core midplane in both tests; the predicted experimental PCT by the code is 677K for ROSA-III and 633K for FIST; the full scale BWR prototype PCT estimated by the code is about 697K and occurs only above the core mid-plane. Figure 4 (a) shows the cladding temperature transients calculated by THYDE-B1 code against ROSA-III and FIST facilities and the BWR prototypical predicted behavior at different axial locations (along the core active region).



(a) (b)

Figure 2. (a) Experimental dispersion band for rod surface temperature at the level 7, (b) correspondent calculated (RELAP5/MOD2) dispersion band [29].



(a) (b)

Figure 3. (a) Lower plenum flashing occurrence *vs. Kv*, and (b) PCT occurrence *vs. Kv*.

In another example [29,30,31] RELAP5/Mod2 code has been used to extrapolate the BWR prototypical behavior. Relevant calculated trends are compared with the counterpart test experimental data showing that (A) the RELAP5/MOD2 code appears suitable in predicting the main phenomena occurring in the three tests; (B) a better agreement with the PIPER-ONE data has been reached by great effort in tuning the nodalization; (C) parameters characterized by different values from each other in the three nodalizations (localized loss coefficient,..) were modified to match the measure trend (no relationship relevant for scaling can be identified with respect to the variation of the tuned values). To calculate full scale prototype behavior two different calculations were performed: BWR-A (BIC as expected in the plant) and BWR-B (BIC scale-up from FIST). As underline in Fig. 4(c), for the rod surface temperature (highest level), the various calculated quantities are similar in a quantitatively and qualitatively point of view. The results of the BWR-B case are similar to those of FIST evolution. The most important discrepancy, from the prototypical calculations, is that no dry out is calculated in the core. This is due to the low value of the overall energy supplied by the core to the fluid during the transient and the value of local pressure drop coefficients fixed in the core without accounting for specific experimental evidence [29, 31].

The comparison between the calculated facilities data and the experimental facilities data have been used (i) to evaluate the capability of the code to simulate the transient at different scales; (ii) to have a feedback on the BWR nodalization considering the tuning of the facility nodalization; (iii) to extrapolate the code accuracy: Y_E/Y_C *vs. Kv*, where Y is a generic quantity relevant for a given transient and the subscript *E* and *C* stay for experimental over calculated. In relation to the accuracy extrapolation, the average accuracy was defined by considering the dispersion of the Y_E/Y_C value around the unit value, Fig 4(b); the resulted accuracy has been applied for the plant calculation, too. By repeating the procedure for all the relevant parameters a realistic "uncertainty band" has been obtained for the code prediction of behavior of the plant. The "calculated dispersion band" correspondent to the related "experimental dispersion band" has been calculated as well. The "best estimate" prediction of the SBLOCA transient in the BWR-6 plant (nominal conditions, BWR-A calculation) was achieved. Figure 2(b) shows the calculated dispersion bands related to the "rod surface temperature at the 7 level" for the FIST/ROSA-III/PIPER-ONE facilities. Figure 4(b) shows the scaling of code accuracy and Fig. 4(c) shows the calculated trend of rod surface temperature at the highest level for the FIST/ROSA-III/PIPER-ONE and BWR.

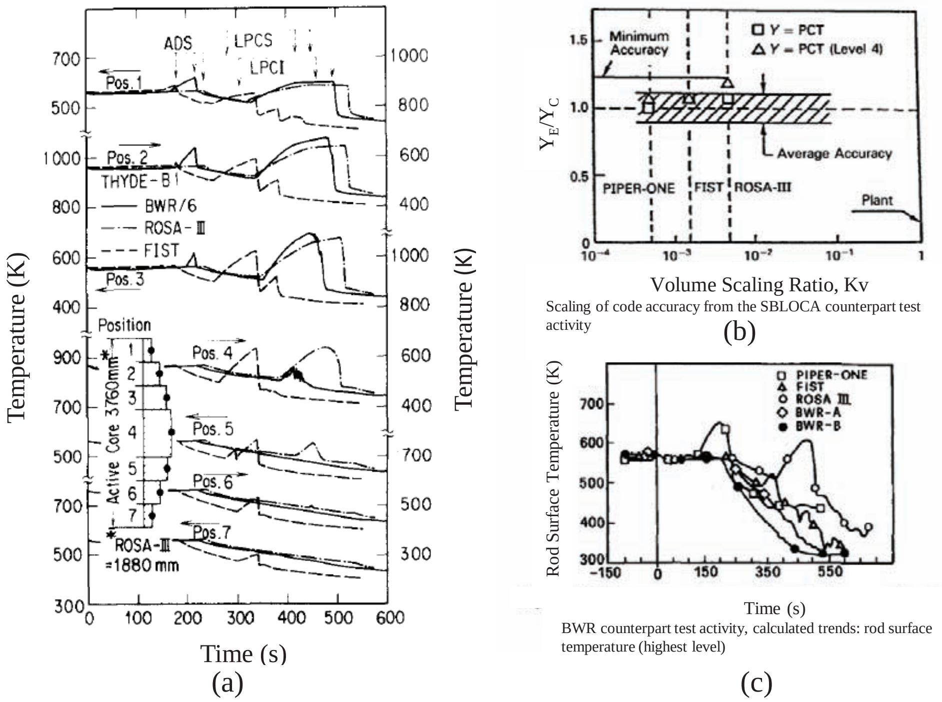


Figure 4. (a) BWR cladding temperature transients calculated by THYDE-B1 code against ROSA-III/FIST counterpart test [28]; (b) Scaling of code accuracy [30] and (c) calculated trend of rod surface temperature at the top elevation for the FIST/ROSA-III/PIPER-ONE and BWR [31].

4. CONCLUSIONS

The unavoidable distortions, characterizing the scaled-down test facility, determine the "facilities scaling-up limits" precluding the possibility of a direct application of the experimental results to the full scale prototype. Therefore, though experimental test facilities and experiments are designed with rationale scaling criteria, obtained results should be limited within the experiments conditions and test scale. In this framework the experimental data developed in counterpart test activities permit to experimentally characterize observed phenomena at different scales; the characterization of the observed phenomena is "restricted" within the experimental range and test facility maximum scale. This is mainly due to the general nonlinear nature of the dominant/relevant phenomena investigated in the facility.

The consequent prediction capability of a "Qualified Code", validated mainly against scaled-down experimental data, is limited within the experiment condition and scale range used for its validation, "code scaling-up issue". The evaluation of the uncertainty in the extrapolated results is one of the most important subjects in Verification & Validation of any computational tool. Within this regard the main issue is how to assure uncertainty, in the calculated results for prototype analyses, without determining the accuracy (in a code validation process) for all the transient and steady physical conditions of interest for a full scale reactor. Therefore each application of a code, outside of its validation ranges, requires caution and should be inserted in a well-defined calculation approach/methodology. It is to underline that SETs to validate code constitutive equations have to range from scale-down facility till full scale to assure the code scaling-up capability.

The “facilities scaling-up limits” and the consequent “code scaling-up issue” determine some considerations about the role of counterpart test data in an assessment database and some considerations about the “judgment of counterpart test data similarity”. In fact since it is not possible to extend the counterpart data at full scale, for unavoidable scaling distortions and the general nonlinear nature of two phase flow phenomena, the “judgment of counterpart test data similarity” is one of the necessary steps (consistency of the database) to apply the experimental data for the assessment of code capability to predict similar phenomena at different scales. Although some phenomenological considerations could be obtained from counterpart test activity (i.e. PCT takes place at the core mid plane and has the same magnitude in ROSA-III/FIST facilities) from the analyses of the ROSA-III/FIST/PIPER-ONE and of the PWR SBLOCA and natural circulation counterpart test [30], it is confirmed that the attempt to have a direct extrapolation (scaling-up) of the experimental data to full scale prototype is not in general possible. In order to scaling-up the “measured transient” to the “expected prototype transient” an extrapolation methodology is always necessary. This is valid for single experimental facility too. The key component of this “extrapolation methodology” is the code.

Though the lack of knowledge related to the full scale plant data, the application of the code inside a well-defined calculation methodology, to assess its accuracy to predict counterpart tests, gives the best answer, up to day, about how extrapolate the facilities results to the prototype. Within this regard the authors think that it will be interesting to investigate if CFD tools in the future can have a role in order to evaluate uncertainty at prototype condition if no experimental data are available. The application of a Qualified code for safety review requires that a further “Safety Analyses Code Adequacy Evaluation” should be satisfied for demonstrating its qualification level. Then the code is ready to be used, for safety review, in a well-defined “Calculation Methodology/Approach” (i.e Conservative, or BE +Uncertainty evaluation).

It to underline that, up to day, the international agreed experimental database status through the publication of the integral test facility validation matrix and separate effect test matrix is updated at the July 1996 and September 1993 respectively. Therefore an internationally agreed update matrix would be necessary in order to include the last 20 years of experiments enlarging the experimental conditions and facility scale investigated.

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