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A state-of-the-art report on scaling in system thermal-hydraulics applications to nuclear reactor safety and design

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– to provide authoritative assessments and to forge common understandings on key issues, as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development.

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The NEA Committee on the Safety of Nuclear Installations (CSNI) is an international committee made up of senior scientists and engineers with broad responsibilities for safety technology and research programmes, as well as representatives from regulatory authorities. It was created in 1973 to develop and co-ordinate the activities of the NEA concerning the technical aspects of the design, construction and operation of nuclear installations insofar as they affect the safety of such installations.

The committee's purpose is to foster international co-operation in nuclear safety among NEA member countries. The main tasks of the CSNI are to exchange technical information and to promote collaboration between research, development, engineering and regulatory organisations; to review operating experience and the state of knowledge on selected topics of nuclear safety technology and safety assessment; to initiate and conduct programmes to overcome discrepancies, develop improvements and reach consensus on technical issues; and to promote the co-ordination of work that serves to maintain competence in nuclear safety matters, including the establishment of joint undertakings.

The priority of the CSNI is on the safety of nuclear installations and the design and construction of new reactors and installations. For advanced reactor designs, the committee provides a forum for improving safety-related knowledge and a vehicle for joint research.

In implementing its programme, the CSNI establishes co-operative mechanisms with the NEA Committee on Nuclear Regulatory Activities (CNRA), which is responsible for issues concerning the regulation, licensing and inspection of nuclear installations with regard to safety. It also co-operates with other NEA Standing Technical Committees, as well as with key international organisations such as the International Atomic Energy Agency (IAEA), on matters of common interest.

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FOREWORD

The word "scaling" has continuously interested the OECD Nuclear Energy Agency (NEA) working groups in the area of thermal hydraulics, i.e. the Principal Working Group (PWG) 2 and the more recent Working Group on Analysis and Management of Actions (WGAMA), since the 1970s. The activities within the framework of the verification and validation (V&V) for system thermal-hydraulics had connections with scaling, e.g. the issue of the state-of-the-art report (SOAR) on thermal hydraulics of emergency core cooling (TECC), the separate-effect test facilities (SETFs) and the integral-effect test (ITF) containment-code validation matrix (CCVM) documents, as well as of several International Standard Problem (ISP) reports between 1980 and 2000. The need to clarify scaling issues was the basis of cooperation among Germany, Japan and the United States within NEA, i.e. the 2D/3-D project involving the Upper-Plenum Test Facility (UPTF), Cylindrical Core Test Facility (CCTF) and Slab Core Test Facility (SCTF) in Germany and Japan. With a similar objective, counterpart tests involving experimental facilities and other programmes of interest within NEA were undertaken by specific CSNI working groups, e.g. LOFT-SEMISCALE, LOFT-LOBI and BETHSY-LOBI-LSTF-PSB-SPES. Scaling also was an important concern in evaluating the uncertainties that are required with the prediction of system thermal-hydraulic codes, a subject that was discussed in the Uncertainty Method Study (UMS) and Best Estimate Methods plus Uncertainty and Sensitivity Evaluation (BEMUSE) projects completed within the framework of the NEA PWG-2 and WGAMA activities.

It also is clear that any validated code or model should be applicable to the conditions from the smallest- to the largest-range scale of any given experiment data. After this validation work, however, the validated code or model is not guaranteed to be applicable to all the nuclear power plant (NPP) phenomena. True validation should provide such a guarantee and shall be part of scaling activities.

Notwithstanding the above activities, a common understanding of the words 'scaling', 'scaling issue' and 'address the scaling issue' was not then available to the scientific- or technological-communities. This need was reflected in the 2012 yearly WGAMA meeting; there was a request for the presentation and for the planning of a possible scaling-related activity. Francesco D'Auria was tasked to formulate a proposal. This involved the following steps:

- Creating a group of proponents/experts: this activity was completed by the end of December 2012.
- Organizing a meeting of the experts: this was held in Pisa on 13-14 June 2013, i.e. the 1st Specialist Scaling Group (SSG) meeting, followed by a statement on scaling from the experts H. Schmidt, D. Bestion, H. Glaeser, H. Nakamura, H-S. Park, O. Zerkak, F. Reventos, P. Lien, and F. D'Auria, representing respectively AREVA, CEA, GRS, JAEA, KAERI, PSI, UPC, US NRC, and UNIPI.
- Formulating a proposal to the WGAMA [2]: this topic was discussed and approved during the 2013 WGAMA meeting, and the decision was to conduct a SOAR on scaling.

- Preparing the CSNI Activity Proposal Sheet (CAPS) for endorsement of the scaling proposal by the Programme review Group and CSNI; the CAPS was endorsed with some modifications by the PRG and CSNI meetings in 2013, see Appendix A-1.
- Starting (formally) the Scaling SOAR (S-SOAR) activity, to produce the present document: the SSG was formed (i.e. the authors of the present document) during the 2nd SSG meeting held in Pisa on Feb. 24-25, 2014.
- Participation, primarily by the Lead Authors, to the following meetings (for completeness, the 1st SGG meeting is included) where the framework, the structure, and the content of the present document was discussed:
  - The 1st SSG meeting, Pisa (I), June 13-14, 2013: the decision to issue the S-SOAR and the proposal for CAPS;
  - The 2nd SSG meeting, Pisa (I), Feb. 24-25, 2014: agreement about structure and content of the S-SOAR and the selection of Lead Authors;
  - The 3rd SSG meeting, Pisa (I), Oct. 21-23, 2014: draft chapters presented and discussion about pending issues;
  - The 4th SSG meeting, Paris (F), Apr. 15-17, 2015: draft S-SOAR available to SSG members;
  - The 5th SSG meeting, Grenoble (F), Nov. 30-Dec. 1, 2015: final version of S-SOAR approved (with amendments) by Lead Authors ('so-called' rev.3 issued on Jan. 10, 2016).

The discussed framework provides the background information on the work performed by the SSG: Emphasis is given to the objective of reaching a common understanding for the concepts 'scaling', 'scaling issue', and 'addressing the scaling issue'.

It is noteworthy that neither the thermal-hydraulic phenomena nor the uncertainty in predictions of the system codes is the main focus of this document. Rather, the role of scaling in the characterization of phenomena and in determination of the uncertainty is discussed.

A glossary is part of the present document. It deals with the scaling-related definitions agreed by the SSG. The glossary also includes a diagram of the role of scaling in different areas of nuclear thermal-hydraulics. In addition to the usual Executive Summary an Extended Executive Summary is included in the present document and constitutes Appendix 5.

The word "licensing" is avoided to prevent possible misinterpretation of discussions and conclusions in this literature. However, licensing requirements are essential in the development of scaling science. Then, the concept of licensing is rephrased as "safety review" or a "safety evaluation" wherever needed in this report.

The following review steps led to this final version of the document: a) 'internal' review by the members of the SSG and by their Organizations; b) independent external review by J-L. Vacher (EDF); c) review by WGAMA members; d) review by the CSNI Programme Review Group members; e) review by CSNI members.

ABSTRACT

The present document deals with scaling in nuclear-system thermal hydraulics (SYS TH), including the connection with Nuclear-reactor Safety Technology (NST). Scaling has constituted 'an issue' since the beginning of the exploitation of nuclear energy for civil purposes, with main reference to the generation of electricity. A nuclear power plant (NPP) constitutes a technologically complex industrial system. There are several origins of its complexity, connected with the need to reduce the cost of producing electricity and to manage the radioactive fission products. This resulted, among the other things in the large pressure vessel, high power and, mainly, high power density (power per unit-core volume), high pressure, and the need for engineered safety-features, including an emergency core-cooling system. Then, another problem was the impossibility of, or the large difficulty in, characterizing the system's performance under the conditions of the design: almost unavoidably, to reduce the cost, the experiments aimed at understanding the original system, here called the prototype, were performed in small-scale systems herein called models. So, models were designed, constructed, and operated under downscaled ranges of values for one or more of the listed parameters. These features lay at the origin of the scaling issue, i.e. the difficulty in demonstrating that a model behaves like the prototype.

Integrated definitions of the widely adopted terms, 'scaling', 'scaling issue', and 'addressing the scaling issue' are part of the present document. The related application domain includes the NST, and the licensing for water-cooled nuclear reactors under operation, under construction, or under an advanced design stage at the time of publication of the document.

Scaling-related analyses are done in different areas of SYS TH and NST. These include the design of test facilities (both integral and separate-effect test facilities, ITF and SETF), the design of experiments (including counterpart test, CT), the demonstration of the capability of any computational tool, and the evaluation of uncertainty affecting the prediction of the same computational tools.

A variety of approaches have been used to address the scaling issue, including non-dimensional analysis of mass, energy- and momentum-balance equations, derivation and application of scaling factors, including the hierarchy of relative importance, performing experiments at different scales, and running the SYS TH computer codes.

This document discusses the key areas and the key approach for scaling. It was found that the SYS TH computer codes, following their application to differently scaled experiments, demonstrate that the accuracy of their predictions may not depend upon the scale of the considered experiments. The TH codes also may constitute an additional valuable tool for addressing the issue of scaling.

The current Abtract shall be seen as the first level of synthesis for the overall document; two additional levels are constituted by the Executive Summary and by the Extended Executive Summary reported as Appendix 5.

EXECUTIVE SUMMARY

Scaling in nuclear-thermal-hydraulics constitutes the topic of this document, completed in 2016 by a Specialist Scaling Group (SSG) formed in 2013 by the Working Group on Analysis and Management of Accidents (WGAMA) of the NEA Committee on the Safety of Nuclear Installations (CSNI). The need for this document testifies the importance of scaling in nuclear technology, but also the controversial evaluations of scaling-related findings by the scientific community. An extended summary of this report is provided in Appendix 5.

As a first priority, a consensus was reached on the terminology. Namely, "scaling" is the process of converting any parameters of the plant at reactor conditions to those either in experiments or in the results of numerical code so to reproduce the dominant prototype phenomena in the model; "scaling issue" indicates the difficulty and complexity of the process, and the variety of connected aspects; and "addressing the scaling issue" is a process of demonstrating the applicability of those actions performed in scaling.

From the impressive amount of research addressing the scaling issue, three categories of activity are identified:

1. Technological bases for scaling, with the experimental data, results of analyses, journal papers, and OECD reports.
2. Requirements for scaling and for the system codes verification and validation (V&V), which include those derived in codes scaling, applicability and uncertainy (CSAU), code with capability of internal assessment of uncertainty (CIAU), best-estimate methods plus uncertainty and sensitivity evaluation (BEMUSE).
3. Scaling techniques and approaches used in scaling analyses, with methods such as power-to-volume, hierarchic two-tiered scaling (H2TS), fractional scaling analysis (FSA), dynamical system scaling (DSS), and the application of system codes.

Chapter 2 discusses the overall "scaling universe" in today's technical community and surveys, as systematically as possible, the commonly-accepted topics and the controversial ones associated with scaling. These topics include scaling distortion, scaling of complex phenomena, and the role of scaling in safety applications and reviews. Starting with an overall picture of the scaling to depict its subjects from several perspectives, some milestone scaling techniques are reviewed and the relationship between thermal-hydraulic scaling and nuclear-reactor safety is introduced. Then, some significant achievements in scaling are highlighted:

– Flashing, flooding and counter-current flow limitation (CCFL) in the downcomer of a pressurized-water reactor (PWR) reactor pressure vessel (RPV) during a large-break loss-of-coolant accident (LBLOCA);
– Wall evaporation, flooding, and CCFL in the downcomer of the steam generator (SG) secondary side, during accident-recovery conditions;
– Influence of reversed-flow SG U-tubes on the natural circulation performance;
– Simulation of nuclear-fuel rods in integral test facilities (ITFs) using electrically heated rods with and without a gap;

- Concept of scaling distortion in the uncertainty method based on extrapolating uncertainty;
- Concept of a Scaling Pyramid that summarizes current scaling approaches.

Scaling Distortion

Identifying and addressing scaling distortions are key issues. Scaling distortions may result from assumptions and simplifications in scaling methods, from technological limitations in constructing and operating test facilities, and from limitations of computer code scalability.

Using the Buckingham Pi theorem or writing conservation laws in non-dimensional form on a selected global or local control volume, a list of non-dimensional groups is generated which define similarity conditions. However, all of them cannot be matched simultaneously in the design of reduced scale test facilities, resulting is some scaling distortion. Some well-known deficiencies identified in the scaling methods are reported.

The details of the scaling methods are described in Chapter 3. The advent of thermal-hydraulic computer codes greatly improved the thermal-hydraulic analyses, a feat that could not be achieved with pure analytical methods. In Chapter 4, the merits and deficiencies of scaling aspects of thermal-hydraulic codes are reviewed in detail. Computer codes also have limited capabilities for simulating reactor conditions. The physical models in the code use empirical correlations for the closure laws of balance equations. Constants in these empirical formulas are sometimes determined by curve fitting, and may depend strongly on the geometry (shape and size), and fluid conditions.

Furthermore, once the code's applicability has been determined, uncertainty in the predicted safety parameters has to be determined and the scaling distortion has to be considered in the uncertainty evaluation process. The relationship of scaling and uncertainty are detailed in Chapter 4.

It is well recognized that distortion is inevitable in scaling complex systems, like light-water reactors (LWRs). Scaling laws usually are derived from the dominant phenomena in each phase of the transient. Since the dominant phenomena may change from one phase to another of the transient, it is unlikely to reach a perfect similitude between the reference system and the experimental model for all phenomena in one transient. The scaling distortion could become large and it is difficult to determine the acceptability criteria for distortion in an experiment. The effects caused by distortions require a method that can evaluate the accumulated distortion of a process as a function of time.

The scaling of complex thermal-hydraulic phenomena is discussed in Chapter 2, including two-phase critical flow (TPCF), entrainment and de-entrainment, core reflooding, fuel-rod ballooning, special plant components such as pumps, separators, and similar ones, core local phenomena at sub-channel level. Among these, TPCF and CCFL are reviewed further in Chapters 3 and 4. Complex phenomena usually affect the operation of the emergency core cooling (ECC) system and its consequences, and cannot be neglected in the scaling. However, they often cannot be described with standard governing equations, and therefore, empirical correlations based on scaled separate-effect test (SET) data were used to derive the scaling laws. This may pose a great challenge to the scaling capability of the models.

In the long nuclear-thermal-hydraulics history, the design of experiments, the construction of the test facilities, the choice of using an integral effect test (IET) or separate effect tests (SETs) or both have been a challenging topic in scaling, which are briefly reviewed. The recent use of counterpart tests and similar tests are also illustrated in Chapter 3.

Scaling Analysis for the Safety-Review Process

In Chapter 2, the scaling role in the safety review process is described with examples of evaluation model development and application process (EMDAP), USNRC, 2005, and quantification of uncertainty, CSAU, USNRC, 1989. Uncertainty approaches are extensively reviewed in Chapter 4, not to recommend any specific approaches, but to illustrate the role of scaling in them.

A safety determination of reactor design and operation is done by evaluating the prototype thermalhydraulic response through data from experiments, and/or computer code calculations. Since a reference reactor cannot provide data for the postulated accidents, simulations of accidents in experiments with scaled test-facilities are inevitable. The scaling technique used to design the test facility is a key element to understanding the validity of experimental data. The core-scaling technology is reviewed in Chapter 3, which covers the scaling methods, and the design of the test facilities and the experiments.

Scaling methods can be categorized by the target phenomena at both the local and system levels. In general, the scaling parameters for a local phenomenon can be derived by applying a dimensional analysis (empirical approach), or dimensionless governing equations (a mechanistic approach). An empirical approach uses correlations and models to derive similarity parameters, or to estimate distortions due to scaling. An example is the criterion for the flow regime transition, based on the Froude number. The approach of the dimensionless governing equation is to simplify the governing equations for both the prototype and model by making assumptions and evaluating the various terms; the similarity criteria can be obtained by comparing the non-dimensional terms in the equations.

To preserve kinematic and dynamic similarities between the prototype and the scaled-down test facility, a scaling method at system level is necessary. Most scaling laws are derived from the non-dimensional governing equations. For ITFs, another level of scaling needs to be completed by preserving the important local phenomena and by reducing scaling distortions as much as possible. The important phenomena and processes can be identified from the phenomena identification and ranking table (PIRT).

Scaling Methods

Each major scaling method is presented with the major characteristics, merits, limitations, and application areas.

1. **Linear scaling –** The same aspect ratio and the same velocity in the model as in the prototype. This approach can excessively distort gravity effects.

2. **Power-to-volume scaling –** This method conserves time and heat flux in the prototype and can reproduce phenomena in which the gravity effect is significant. It is suitable to simulate an accident in which flashing occurs during depressurization. It was successfully used to design most of the integral-effect test facilities, such as LOFT, SEMISCALE, LOBI, ROSA-II, ROSA-III, PKL, LSTF, and BETHSY. Also, this method is suitable for the heat-transfer test with electric fuel bundles. However, when it is applied to a smaller facility with the full height, due to the smaller area ratio, some important phenomena can be distorted, for example, the excessive stored heat in structures, a higher-surface-to volume ratio leading to higher heat losses from structures, and distorted multidimensional flow phenomena.

3. **Three-level scaling –** The first step is an integral - or a global- scaling analysis to conserve a single and/or a two-phase natural circulation flow, using a 1-D non-dimensional governing equation of natural circulation. The second step is a boundary flow and inventory scaling. The geometry is specified to scale the flow rate at the junction of a broken part, the safety-injection system, and various filling- or discharge-systems in the ITF to ensure the similarity of inventory of the mass and energy is preserved in the ITF as a model of the prototype. In the last step, a local phenomenon scaling is performed to conserve the important thermal-hydraulic phenomena occurring in each system. The result from scaling the local phenomenon takes priority if the similarity requirement differs from that derived in the integral scaling. The three-level scaling method is characterized by relaxing restriction on the length scale. By adopting a proper length-scale, some distortion of the flow regime and multi-dimensional scaling in the scaled ITF can be reduced. On the other hand, the scales

for time and velocity are lowered due to the reduced length. Consequently, some local phenomena could be distorted.

4. **Hierarchical 2-Tiered Scaling (H2TS) –** The procedure consists of four stages, i.e. system decomposition, scale identification, top-down analysis, and bottom-up analysis. At the first stage, the system conceptually is decomposed into subsystems, modules, constituents, phases, geometric configurations, fields, and processes. The scale identification as the 2nd stage provides the hierarchy for the characteristic volume fraction, spatial scale, and temporal scale. To establish the hierarchy of the temporal scale, the characteristic frequency of a specific process is defined, and then the characteristic time ratio can be found by dividing by the system's response time based on a volumetric flow-rate. The top-down scaling as in the 3rd stage offers a scaling hierarchy, using the conservation equations of the mass, momentum, and energy in a control volume. In the non-dimensional balance equations, the characteristic time ratio represents a specific transfer-process between constituents. All the processes can be compared, and ranked for importance on the system to establish priority in the scaled models. The bottom-up scaling, as the 4th stage of the method, offers a detailed scaling analysis for key local phenomena, such as the CCFL and choking. Along with this top-down analysis, similarity groups (called Pi groups) are identified, and the scaling criteria and time constants can be obtained to evaluate the relative importance of the processes.

5. **Power to Mass Scaling –** To determine the test conditions for a reduced-height and reduced pressure (RHRP) facility, the power-to-mass scaling method was developed. This method determines scaled core-power according to the initial coolant's mass inventory in the reactor's coolant system. The temperature of the hot leg in the test facility is determined from the subcooling of the primary system, which is made the same between the model and the prototype. The cold-leg temperature is determined by the equivalence of the core temperature difference for the model and prototype. The mass flow rate of the core is scaled down according to the power and heat capacity relationship. Finally, secondary system pressure is determined from the difference in temperature between the primary- and the secondary-side. Since pressure is not preserved, the differences in fluid's thermal properties could induce distortions.

6. **Modified Linear Scaling –** The multi-dimensional behaviours of the Emergency Core Cooling (ECC) water in the downcomer (e.g. the ECC bypass) are observed during the LBLOCA refill phase. The modified linear scaling method was developed to overcome this distortion in a small-scale test facility. Twelve dimensionless parameters were obtained from the two-fluid momentum equations in the downcomer. By preserving those parameters in the model, the method resulted in the same geometric similarity criteria as in the linear-scaling method. However, this method conserves the gravity scale. It also was found that the three-level scaling method provides the same requirements when the area aspect ratio is preserved as square of linear ratio in a test facility.

7. **Fractional Scaling Analysis (FSA) –** FSA is a hierarchic approach similar to H2TS. In the first step, the regions of interest and the durations of the transients are specified. The rate of change of the state variables over the region are connected to the transfer functions defined at the boundary, and inside the volume. The relative effect of components is based on their relative impact on state variables in the transfer function connected to that component. These relative values determine the importance of these transfer terms. The fractional change- of-state variable (effect metrics) over the characteristic time (fractional change metric) should be made the same between the prototype and its model in top-level scaling. The characteristic time is obtained either from the experiments, or from an aggregate fractional rate of change (FRC, also called the aggregate frequency). The individual FRC can be positive or negative. The reference value of the agent-of-change should be the maximum value over the period of the phase. FSA offers a systematic method of ranking components and their phenomena in terms of their effect on the figure of merit (FOM), or the safety parameter. It also can estimate scale distortions, and synthesize data from different facilities for the same class of transients. This multistage scaling can guide the design, and simplify the scaled facility by identifying important components and corresponding processes. This approach does not require the preservation of time.

8. **Dynamical System Scaling (DSS) –** To address the time dependency of scaling distortion, an innovative approach was developed recently converting the transport equations into a process space through a coordinate transformation, and exploit the principle of covariance to derive similarity relationships between the prototype and model. After the transformation, the target process can be expressed in the process-time space as a three dimensional phase curve, called geodesics. If a similarity is established between model and prototype, these two phase-curves will overlap at any moment of the transient. Any deviation of the process curves represents the deviation of scaling as a function of time. By specifying the ratios of the conserved quantity and the process (called 2-parameter transform), the generalized framework can be converted to a specific scaling method, such as the power-to-volume scaling. Furthermore, this generalized approach offers the benefit of identifying the distortion objectively and quantitatively at any moment of the transient.

Depending on the objectives of an experiment, as well as the budget and facility building size constraints, the approach is applied for scaling height (volume), time and/or pressure. One criterion is to maintain the minimum dimensions to preclude some size effects such as surface tension effects that would not occur in the prototype. A dimensionless diameter was established which must be greater than approximately 32-40 to preclude the influences of surface tension. Other criteria related to hydraulic resistance (friction numbers), stored heat, and heat loss need also to be considered.

Scaling methods are essential tools in the nuclear thermal-hydraulics, but they are not sufficient to address the needs in quantifying the safety margins. Limitations of the methods are related to the choice of starting equations, to approximations made in evaluating non-dimensional numbers, to details of geometry and of the initial conditions of the NPP, to the local validity of scaling criteria.

The experiments are indispensable to complement the scaling methods to address the safety margins and uncertainties in the safety of nuclear reactors.

Role of Experiments in Scaling

The experiments in nuclear thermal-hydraulics can be grouped into three categories: basic tests, Separate-Effect Tests (SETs), and Integral-Effect Tests (IETs). Basic tests aim at understanding the phenomena, do not make necessarily reference to the geometry nor to the actual ranges of operating parameters in power plants; they have a weak connection with scaling.

SETs are designed to observe phenomena in selected zones in a nuclear-power-plant's system or in specific plant components and some specific process in a particular period of a given transient. The major role of SETs is to provide experimental data to develop and validate the physical models, and/or empirical correlations under prototypical or simulated-conditions. Recently, heavily instrumented SET facitlities (SETFs) were built to produce spatially and temporally fine-resolution data for validating the computational fluid dynamics (CFD) codes (called CFD-grade experiments).

The IETs use Integral Test Facilities (ITF). An IET provides a similar thermal-hydraulic dynamic response to a postulated accident, and/or abnormal transient in a reference reactor. The data obtained from scaled ITF experiments are considered not directly applicable to full-scale conditions due to scale distortions. They are mostly used for understanding accident phenomena and validating the system codes.

A comprehensive appendix summarizes the key parameters of the major IET test facilities in the world, including ITFs for PWR, boiling-water reactor (BWR), vodo-vodyanoi energetichesky reaktor (VVER), advanced reactor and containment, and selected SETFs. The focus is on how scaling has been considered in the design and the experiment results. Scaling methods were applied to advanced design features, such as passive systems, interactions between the containment and reactor coolant system (RCS), and low pressure phenomena.

The SETF for reactor systems often have minimum scaling distortions by employing full-scale and/or prototype fluid conditions. They have well known boundary conditions and use dedicated instrumentation to characterize selected phenomena. In many cases, the data obtained from a SETF can be applied to the full-scale prototype. However some scaling distortions may be due to boundary conditions, to the facility's scale, or to non-preserved 2D and 3D effects. Therefore, a full-scale SETF such as the Upper-Plenum Test Facility (UPTF) is valuable to characterize multi-D phenomena. In addition, counterpart tests for the same phenomenon also provide confidence in extrapolating to a full-scale plant.

For the SETF of the primary containment vessel (PCV), the present state-of-the-art report (SOAR) focused on some highlights on the scaling techniques related to the design-basis accident (DBA) phenomena. Different containment designs for PWRs, boiling-water reactors (BWRs), water-water energy reactors (VVERs), and small modular reactors (SMRs) are compared. For the PWR PCV-ITF, earlier facilities were a part of small yet real power plant. The interest on scaling arose when discrepancies were observed in the results between HDR (Heissdampfreaktor) and BFC (Battelle-Frankfurt Containment). The facility's material, the compartmental subdivisions and the energy- release are important in designing PCV-ITF.

The scaling compromise is one of the major causes of scaling distortions due to the difficulty of achieving complete similitude in all local phenomena and also due to the lack of knowledge of the local phenomena. In this review, the following main scaling distortions observed in the experiment are identified and are as follows:

1. Circular sections with reduced hydraulic diameters not preserving friction and heat losses.
2. Overestimated structural stored heat and surface area per unit coolant volume.
3. Inventories and inter-component flows with possible choked flow.
4. Pressure drop with too large length to diameter ratio.
5. Multi-dimensional phenomena – due to tall and narrow nature to preserve power volume and height.
6. Scaled-down reactor coolant pump – reliable two-phase pump model is not available until now; specific speeds and single-phase characteristics are recommended to be preserved.
7. Fuel simulators – electrically heated fuel simulators may behave differently from nuclear fuel rods.
8. Scaling distortions of local phenomena – due to inherent scaling distortions by design and simulation constraints, and non-typicality of local phenomena.

It should be noted that not all local phenomena are of equal importance in influencing the FOM or the parameter of interest. The global scaling approach provides that guidance.

Counterpart Test (CT) and Similar Test (ST)

As data acquired in experiments at a single (scaled) test facility may be questionable due to inherent scaling distortions, the concept of counterpart tests (CTs) involving several ITFs or SETFs at different scales and design approaches, have been considered important. It is desirable that the following minimum set of BC/IC and parameters are preserved between the CTs.

1. Thermal-hydraulic state and parameters (pressure, temperature, and flow condition) in each component of the facility.
2. Scaled values to power-to-volume scaling ratio (kv).
3. Characteristics of primary- and secondary-side safety and operational systems (e.g. accumulator injection and safety-injection systems (SIS) characteristics).
4. Heat- and mass-sinks or sources (e.g. location and size of break).
5. Timing of operator's actions based on pre-defined operational criteria.

Good examples of CT include the small-break LOCA (SBLOCA) tests by LOBI, SPES, PSB, BETHSY, and LSTF. All five test facilities simulate the primary circuit of a Western PWR (VVER in the case of PSB) with original heights covering a broad range of volume-scaling factors from 1:712 (LOBI) to 1:48 LSTF). The similarity of the overall results confirms the choice of the adopted scaling laws and the suitability of the individual test facilities to reproduce a plant's typical behaviour under the given BCs. The CT tests conducted within the NEA PKL-2 and ROSA-2 projects demonstrated the effectiveness of a secondary-side depressurization in removing the heat from the primary side, and achieved almost identical primary-depressurization behaviours. ITF tests whose boundary and/or intial conditions (BC/IC) were not aligned according to the requirements of CT are referred to as "similar tests" (ST). A special group of tests called complementary tests where, in the same set up, the ITF concentrates on studying the overall system's response and the SETs investigate the responses of the plant's subsystems and phenomena which are highly dependent on the geometry (in scales up to 1:1 full-scale, such as UPTF).

Another category of tests referred to as daughter (facility) tests compares results available in 1:1 full-scale as the reference with the results from scaled-down experiments on the same phenomena. It aims at evaluating the scalability of relevant phenomena and their understanding in general.

Role and Characteristics of the System Code

System codes incorporate the knowledge obtained from the available large data base. Mature system codes can then assist PIRT and scaling analyses. The merits and limits of codes related to scaling are reviewed.

In the process of developing code, several averaging simplifications are made on the space- and time-scale of the processes. Some distortions are introduced due to simplifications of the physics, non-modeled phenomena, and the limited accuracy of the closure laws. Therefore, several inherent limits are summarized here:

– **Space and time averaging**: System codes do not predict small-scale thermal-hydraulic phenomena due to space averaging and cannot predict all the small time-scales associated with turbulence and two-phase intermittency.
– **The dimensions of the model**: Using the O-D (or lumped) model, 1-D models, or a porous 3-D approach consists of simplifying a complex 3D flow; using a 1-D heat conduction in heating structures and in passive solid structures as an approximation for more complex 3-D conduction.
– **Flow regime maps**: The highly empirical flow-regime maps are valid only in steady state or quasi-steady state, in fully developed or quasi-developed states, while the rapid transient- and non-established-flows could exist in accident conditions. The flow regime should also depend on geometry, conduit size, and the fluid's physical properties but information is missing on all these effects
– **Scaling of each closure law:** Closure laws in system codes may be either purely empirical, mechanistic, or semi-empirical. Therefore, the scalability of some closure laws is questionable.
– **Non-modelled phenomena:** System codes neglect many complex phenomena. Hence, the up-scaling capabilities of a system code depend mainly on how well it predicts phenomena in scaled SETs and IETs.

The scalability needs to be confirmed during the process of validation and, to a less extent, during the verification. The code validation with various scaled SETs and/or IETs plays a very important role to assure the scalability of the code. The scalability of each closure law may be checked using SETs. When phenomena are distorted in scaled IETs, the code scalability needs validation against the same phenomena in non-distorted SETs.

Scaling should be considered in developing the nodalization. For instance, it is impossible or impractical to preserve the length-to-diameter ratio (L/D) when setting up nodalization for differently scaled facilities. Choosing a reasonable size of the control volume is important for acceptable numerical solutions. A specific scaling qualification is needed for the K-factors (local pressure-loss coefficients) at geometric discontinuities in a nodalization. An analyst should follow procedures and criteria to pass the nodalization from the scaled facility to a NPP, and to guarantee that the uncertainty derived in scaled facility remains acceptable under NPP conditions. An approach, called Kv-scaling, is a procedure for system code simulation in which well-defined (measured) scaled ITF- are converted to an NPP-nodalization, and the test is simulated with this nodalization. The purpose is to reproduce, by sensitivity studies, same phenomena as seen in ITF by the NPP nodalization. Performance of both NPP and ITF nodalization can be compared to check the validity of the NPP nodalization for any needed corrections and improvements. The procedure is systemized to qualify NPP nodalization.

The system code can be used in the preliminary verification of the scaling laws. To study distortions, Ransom et al., 1998, devised a triad method, somewhat reflecting the Kv-scaled method, to relate the scaled experiment to the prototype system. The method is based on three separate, but related system-code models: (1) The prototype; (2) an ideally scaled model; and, (3) the actual scaled experiment. These three models are created to investigate the degree to which qualitative- and quantitative-similarities are maintained among the three systems in a particular process. The triad method ensures the qualitative- and quantitative-similarity of the response of the prototype and the ideally scaled model and shows the effect of distortions due to any non-typicality, heat loss, real valves' opening times.

Scaling in Uncertainty Methods

The relationship of scaling and the uncertainty method is another important subject since scaling is also a source of uncertainty in the prediction of NPP transient. Three uncertainty methods – CSAU, uncertainty methodology based on accuracy extrapolation (UMAE)-CIAU, and the GRS Method are reviewed.

In the CSAU procedure, three uncertainty sources are quantified as follows: (a) The code and experiment accuracy, (b) the effect of scaling, and, (c) the reactor's input parameters and state. The first two are normally combined. Using information from the PIRT results and the code assessment manual, uncertainties and biases are determined based on the following two sources as:

1. Evaluation of scaling distortion of a phenomenon in test facilities at various scales;
2. Evaluation of scale-up capabilities of closure laws used in the code.

All available scaled data used to develop the correlation or model in the code are compiled to determine the uncertainty or bias so to reach the 95% confidence level. Additional biases are needed if the range of NPP conditions is not covered in the tests. After evaluation, all the uncertainties and biases are added together as the total uncertainty in the FOM.

In UMAE, experimental data is related to the corresponding calculated results, and an 'error-scaling' procedure is performed. Therein a database is constituted by time trends of the relevant thermal-hydraulic parameters measured in ITFs with different scales and their 'qualified' code calculations. As some conditions are met, e.g. a sufficient number of experiments in different scales and the error of prediction is not scale-dependent, then the error which shows a random character can be extrapolated to the NPP' conditions. A key scaling step of UMAE is the similarity between the NPP prediction and one set of ITF experimental data. This state is achieved through the Kv-scaled calculation.

The GRS method is a widely used uncertainty method based on probability calculus and statistics. The main advantage in using these tools is that the number of calculations is independent of the number of uncertain parameters to be considered. The necessary number of code calculations is given by the Wilks' formula, which depends only on the chosen tolerance limits, or the intervals of the uncertainty statements of the results. The method requires first identifying the important phenomena (PIRT), and then the potentially important contributors to the uncertainty of the code results. Uncertainty due to scale effects is one of them. The probability distributions of each phenomenon uncertainty must be quantified. After qualification process is done for code, and the nodalization is established, the combination and propagation of uncertainties is executed. Finally, the scale-up effects in the method are evaluated by quantifying model uncertainties in facilities of different scales and uncertainties due to input.

Scaling Roadmaps

Scaling roadmaps are discussed which focus on the design of experimental facilities, and on the nuclear reactor's safety assessment. One of the scaling roadmap for designing test facilities is based on the DSS method already discussed.

Address scaling issues in a safety-review process uses the available data, tools, methods, and approaches. A scaling roadmap is proposed to group these actions and information. Due to the different BEPU approaches, there are different ways to meet the safety requirements. Two scaling roadmaps are provided for the reader's reference.

A generic scaling roadmap is proposed, first based on CSAU with a scaling method chosen to design test facilities. These test facilities provide essential information for designing the plant, and for assessing the efficacy of safety systems. With the data, the expected thermal-hydraulic processes and phenomena of power plant can be simulated through calculations with the system code. The results obtained are evaluated by regulators. The fidelity of predictions is estimated by aggregating the contributions of uncertainties from the code models, nodalization, numerics, user options, and approximations in the power plant's representation.

Another scaling roadmap, proposed by D'Auria & Galassi, 2010, is also described. In this approach, most elements in the Scaling Database and Knowledge Management constitute the major steps. Differences from the previous roadmap are that some qualitative- and quantitative-acceptability thresholds are embedded in the major steps. These safety requirements either are established by the regulator or first proposed by the licensee and accepted later by the regulator. Non-compliance of safety requirements leads to halting of the procedure and requesting for additional calculations, experiments, and/or R & D.

Role of CFD Tools for Multi-dimensional and Multi-scale Phenomena

3D CFD tools become valuable when multi-dimensional effects play an important role in issues such as single-phase turbulent mixing problems, including temperature mixing, mixing of chemical components in a multi-component mixture (boron in water, hydrogen in gas) and temperature (density) stratification. Two-phase CFD is much less mature than single-phase CFD, but significant progress has been made in the past decade. Two different 3-D simulation approaches can be used in reactor thermalhydraulics for design, safety and operation studies:

1. CFD in porous medium: This approach is dedicated to design, safety, and operation studies for reactor cores, heat-exchangers, and to the pressure vessel. Each mesh or control volume may contain both fluid and solid structures. The minimum spatial resolution is fixed by the hydraulic diameter, i.e. the sub-channel's size (scale in centimeters) in a sub-channel analysis.
2. CFD in open medium: The space and/or time resolution is smaller than in the previous approach. It includes turbulence modelling, using either the Reynolds-averaged Navier–Stokes (RANS) approach or large-eddy simulation (LES). It also is the only scale that, in principle, can predict the fluid temperature-field, thermal shocks, or thermal fatigue.

Conclusions

The purpose of a state-of-the-art review is to survey the status of scaling technology from different perspectives. However, the technology continues to evolve, and new methods and approaches are being developed. Therefore, it is not appropriate to draw specific conclusions. A few broad conclusions are summarized as follows:

1. The information in scaling studies, namely the experimental database, is available for most

reactor types but has not been fully exploited.

2. Scaling methods and models are available for specific targets or objectives. The application to a generic objective may suffer from the limitations of these methods
3. Many non-dimensional scaling groups are derived in scaling methods and models: knowing the hierarchy of these groups is important in applying scaling methods.
4. Distortions cannot be avoided in any reduced-scale experiment where transient two phase flow is involved. Even in the case of single-phase conditions phenomena, like stratification and entrance effect, may induce distortions in scaling, particularly in passive systems.
5. The impact of scaling distortions upon the performance predicted for any reference system, prototype, or reactor, remains difficult to quantify.
6. Data from scaled experiments cannot be directly extrapolated to the reactor in most cases dealing with two- phase flow.
7. Use of a suitable existing scaling method or development of a new method for a specific experiment is essential in minimizing scaling distortions.
8. The use of a well validated and verified SYS TH code can support any scaling analysis, including checking the scaling hierarchy, evaluating the impact of scale distortions, and correcting the distortions in reactor applications. For a safety determination of an NPP, the application of SYS TH codes can support, but not replace the formal scaling analysis, and is the best tool for up-scaling to the reactor transient of interest after the two following requirements are met (i.e. items 9 and 10 below).
9. Uncertainty from scaling should be accounted for in the overall uncertainty when the SYS TH code is used in predicting the thermal-hydraulic phenomena in NPP accident scenarios.
10. Accurate evaluations of scaling uncertainty in the validation results, model correlations, numerical schemes, and nodalizations are needed to meet the requirements of nuclear reactor safety.

Recommendations

Based on the key findings in each chapter, the recommendations are summarized here without prioritization, for planning future activities.

1. To resolve a safety issue related to a postulated reactor accident, the most reliable approach should combine the use of PIRT analysis, scaling analysis, analysis of a wide SET and IET experimental database (including counterpart tests), and the use of a system code in a best-estimate plus uncertainty (BEPU) approach. In some cases a multiscale simulation using CFD tools may provide better insights into local 3-D phenomena.
2. The capability of SYS TH codes to predict facilities of different scales is needed to evaluate the safety of light water reactors (LWR). The recommendation is to include the scalability requirements in SYS TH code validation. The counterpart tests will also be important asset for validating scalability of the codes.
3. The database of existing SETF and ITF computer-code validation matrix (CCVM) should be extended to include possibly data related to advanced reactors (including those using passive safety systems), radial transfers due to diffusion, dispersion of momentum and energy, and cross flows in the core.
4. There is a need for well instrumented tests for validating CFD codes for the water cooled reactors in relation to mixing problems, such as boron dilution, main-steam-line break (MSLB), pressurized thermal shock (PTS), thermal fatigue, or mixing with buoyancy effects in some passive systems, to be considered in the general TH validation matrices. CFD codes must first be validated on single phase tests at different scales.
5. There is a need to identify a qualitative and quantitative framework (precision targets) to judge

the quality of a scaling approach. This step is connected with the acceptance criterion for scaling distortions, and with the quantification of uncertainty due to scaling.

6. Full height scaling with suitable flow areas (and volume) are recommended for experimental simulation of passive system, wherein the important phenomena are the boiling and condensation processes, and buoyancy effect due to density change. Full height will provide an accurate characterization of phenomena such as natural circulation and related stability.
7. Specific scaling related training is worthwhile in a number of contexts. On both the industry and regulatory sides, good training and education of safety analysts should include, in addition to basic single phase and two-phase thermal-hydraulics, advanced topics of scaling techniques, identification of the dominant phenomena of major transients, code verification and validation (V&V) and uncertainty quantification (UQ) requirements, and code scalability requirements.
8. Revisiting systematically the scalability of system codes at the basic level of each closure law may be a good exercise for training new code users, so to improve the understanding of code scaling uncertainty and to improve code documentation.

Multiscale analysis using several numerical tools at different scales will help in future to provide more accurate and reliable solutions to reactor issues. This approach requires first that the capabilities and limitations of 3-D two-phase flow calculation (CFD) methods for flows relevant to an NPP are well identified.

The simulation capability of details of local phenomena aiming for a replica of the phenomena must be improved. Up-scaling methods for modelling should be developed to use small-scale simulations for improving the closure laws used in SYS TH codes. The CFD tools also should follow an appropriate process of code validation to prove their capability for extrapolation to the NPP-prototype phenomena.

LIST OF FIGURES

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LIST OF TABLES

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1. INTRODUCTION

Scaling is a reference 'key-word' in engineering and in physics: Scaling constitutes a universal problem for many technologies. Its relevance in nuclear-reactor technology constitutes the key motivation for the present document.

The size and the complexity of the nuclear reactors, including the operating conditions connected with the need to optimize cost and safety is at the origin of scaling-related problems. Industry uses plants of large geometric size, coupled with high pressure and high power to produce electricity at reasonable cost: this makes it impractical to perform experiments with the same size, pressure, and power. The experiments are undertaken when either one or more of the parameters that characterize the geometry, the pressure and the power, are smaller than those in the original system. Hereafter, the original and the scaled-down systems are termed as prototype and model, respectively

The applicability of the data measured in the models to the conditions expected in the prototype is the origin of the terms 'scaling', 'scaling issue' and 'addressing the scaling issue'.

1.1 Background and Scope

Generic Background and Scope

The generic technological background for the present document comprises the terms 'scaling', 'scaling issue' and 'addressing the scaling issue' that already were introduced and defined - in the Glossary - as follows:

> 'Scaling', 'scaling issue' and 'addressing the scaling issue' indicate the actions, the methods and the approaches aimed at connecting the parameter values related to experiments with Nuclear Power Plant (NPP) conditions; the subject parameter values are applicable and qualified under the reduced-scale conditions; the reduced-scale conditions imply values of geometry, pressure, or power, or combinations, smaller than the values characterizing the NPP conditions. Scaling is the process of converting any plant parameters at reactor conditions to those either in experiments or in numerical code results in order to reproduce the dominant prototype phenomena in the model. Scaling issue indicates the difficulty and complexity of the process and the variety of connected aspects. Addressing the scaling issue refers to a process of demonstrating the applicability of those actions performed in scaling.

The scaling-issue arises from the impossibility of obtaining transient data from the prototype system under off-nominal conditions. Solving the scaling issue implies developing approaches, procedures, and data suitable for predicting the prototype's performance utilizing small-scale models.

The scope of this document is restricted to reactors that use water as coolant and/or moderator (this means all the reactors currently producing electricity), although the concepts and procedures can be applied to different types of reactor. Within the NPP technology, prototype data is available from nominal operating conditions. So, the key interest in scaling is for nuclear- reactor safety technology (NST) wherein prototype transient data are not available.

The NST makes use of approaches, procedures, and data for a broad variety of situations, both with the high- and low-probability, expected during the life of a reactor. However, the requirements and our technological understanding may be substantially different for events with high- and low-probability. A key technological boundary separating the two events constitutes the envelope of the design-basis accident (DBA). The boundary is defined under the situation of major core-damage with the loss of geometric integrity, i.e. the core condition depicted in beyond design base accidents (BDBA). The difference between these two areas (i.e. DBA and BDBA) is reflected in different requirements and acceptance criteria, in investments of R & D in the two areas, in the quality- and qualification-levels for the tools used in predicting the off-nominal conditions.

Thus, the scope for this document is restricted to DBAs that occur before the loss of the core's geometric integrity. A consistent pilot activity was performed (see the Foreword) to plan the present document, D'Auria, 2013, D'Auria et al., 2013, and NEA/CSNI/WGAMA, 2013. The related reports afford additional background and references for the present activity. This especially is true in relation to the commitments taken by the Specialist Scaling Group (SSG) members with the NEA/CSNI WGAMA CSNI activity proposal sheet (CAPS) given in Appendix A-1.

NST Background

Apparently, the terms 'scaling', 'scaling issue' and 'addressing the scaling issue' are the center of attention of the scientific community within NST (key references are given in the section below). The word 'scaling-controversy' sometimes is used to depict the status of our current understanding about scaling. It is the motivation of the current project discussed in section 1.2.

Database of Scaling Knowledge

Selected elements of scaling, applicable to the scope characterized above, are synthesized in Fig. 1-1, i.e. the Scaling Knowledge Database and Management. These elements are categorized into three groups, other than scaling achievements that are discussed separately:

⇨ Category 1: The technological bases for performing scaling.

The knowledge acquired in developing SYS TH codes over the years, spread within and agreed upon by the international community, belongs to this category. This includes the CSNI SOAR on TECC, NEA/CSNI, 1989; the reports documenting the design, the test matrix, and the analysis of experiments in scaling significant ITF, e.g. NEA, 1991, Addabbo & Annunziato, 2012, Bazin et al., 1990, and NEA, 2007; the reports documenting the design, the test matrix, and the analysis of experiments in scaling significant SETF, e.g. USNRC, 1993, and EPRI, 1982; the Compendium on ECC research issued by US NRC as the back end of huge investments, USNRC, 1988; the CSNI CCVM on ITF, NEA/CSNI, 1987, and NEA/CSNI, 1996, and on SETF, NEA/CSNI, 1993; the special issue of the J. NED issued in 1998, devoted to scaling.. The contents of the listed documents are deemed fundamental and introductory to any scaling study; therefore, they are shown as the foundation of the scaling-knowledge management. This also encompasses many system TH codes for NPP simulation, plant data, and knowledge gained from analysing transients.

Noticeably, the referenced documents deal with experimental data that are needed for the development of SYS TH codes. These experiments have also been designed based on scaling knowledge and have been used to solve scaling issues, and to identify and sometimes to characterize phenomena expected in nuclear reactors.

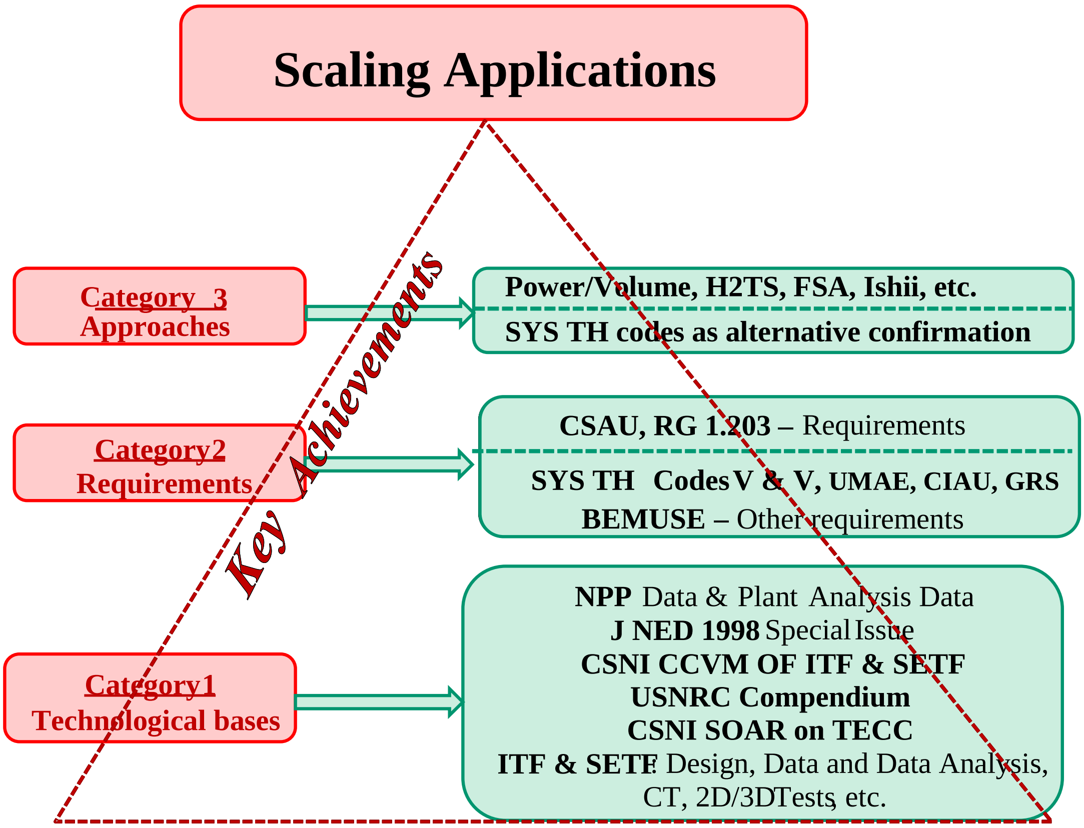


Fig. 1-1 – Knowledge management for scaling.

⇨ Category 2: The requirements for scaling.

There are no universally accepted requirements for scaling analysis in the public domain. However, the pioneering effort by the CSAU methodology and its related documents, USNRC, 1989, touched upon the requirements of scaling. CSAU methodology often is viewed as a basket containing many requirements rather than a prescriptive methodology to guide uncertainty analyses (see Chapter 4). CSAU was established based on the studies referenced in Category 1, with the exception of NED Special Issue, NED, 1998, which was published a decade after CSAU (as a feedback of the CSAU requirements).

The requirements from CSAU are better interpreted as targets for scaling analysis to be achieved, based on the available knowledge.

In December 2005, the USNRC published Regulatory Guide 1.203 – Evaluation Models Development and Assessment Procedure (EMDAP), USNRC, 2005. This regulatory guide is intended to provide guidance for developing and assessing EMs for accident- and transient-analyses. EMDAP is a multiple-step procedure. In the Step 6, licensees are expected to provide a scaling analysis, and to identify similarity criteria. And in Step 8, licensees are expected to evaluate effects of IET distortion, and the capability of scaling up SETs (more details provided in section 2.4.1 and in Chapter 4).

Code validation is an important step in establishing code capabilities and requires well-scaled tests representing the phenomena in the codes. The V & V process has the capability to characterize errors in System TH code predictions, and therefore, the need for an uncertainty estimate. The uncertainty evaluation methods, NEA/CSNI, 2006 and IAEA, 2010, require the resolution of scaling issues. The uncertainty methodologies, UMAE and CIAU, D'Auria et al., 1995, and D'Auria & Giannotti, 2000, directly use scaled data and extrapolation of accuracy.

The CSNI BEMUSE project, e.g. NEA/CSNI, 2006, and the IAEA documents that deal with the approaches required analysing accidents, e.g. IAEA 2002, IAEA 2008, and IAEA 2010, should be part of the current category. Namely, the CSNI BEMUSE project attempted to connect the scaling and the uncertainty of code predictions (see Chapter 4). Uncertainty analysis was performed by a group of experts on a scaled model (the LOFT experimental facility) with available experimental data (noticeably a double-ended guillotine-break LOCA). The same group also undertook uncertainty analysis towards the prototype (industrial NPP in relation to which suitable information to perform analyses was available), so trying to transfer the results from the former analysis to the latter, and to estimate any scale distortion.

The IAEA documents are related to the BEPU approach and NPP-safety requirements with which that the scaling technology of the NPP designers needs to comply. BEPU, unlike the traditional conservative approach, is an approach to performing safety analysis using the best available techniques. Adopting the BEPU approach requires addressing the scaling issue, e.g. the code's scalability and uncertainty.

The different nature of requirements established by the CSAU and by other listed documents should be noted. In the former case, direct targets for scaling analyses are proposed. In the latter the requirements shall be derived from, and are related to the adopted procedures; these shall be seen as common practices accepted by the scientific community.

⇨ Category 3: Techniques and approaches for scaling activities.

The methods of scaling analysis have the objective of pursuing scaling targets based on the knowledge available in documents in Categories 1 and 2 (see also the summary table in Appendix 5).

The general approaches of scaling analyses are based on conservation laws. Approximate methods are used to calculate the reference's quantities to accomplish analyses. An exhaustive evaluation of methods is not provided, rather, significant examples are given.

The first approach includes contributions by individuals or groups of experts: the papers published by the J NED 1998, Special Issue, are part of this sub-category. Significant examples are given below but not in order of their importance.

The Westinghouse comprehensive scaling analysis for the IRIS (design stopped), a small size reactor, is an example for the first pillar, Dzodzo, 2014. The pioneering work by Navahandi et al., 1979, constitutes a reference example of scaling methods, Important scaling activities were undertaken by Novak Zuber: namely, the universally accepted concept of hierarchical approach to scaling, that is part of the H2TS, Zuber, 1991, and acceptable concept of identifying phenomena within an accident scenario and characterizing corresponding scaling groups is part of the FSA, Zuber et al., 2007. Furthermore, pursuing FSA and having available suitable experiments is an alternative way to undertake scaling analysis without the help of SYS TH codes. The category of scaling methods and approaches with which we are concerned includes for example the database of Natural Circulation (NC) in PWR, D'Auria & Frogheri, 2002. In this case a variety of experiments, performed at different scales, were used to create a NC (bounding) flow map that proved useful for interpreting the performance of systems, different from those which originated the map. The Counterpart and the Similar Tests, e.g. Blinkov et al., 2005, constitute a powerful (and expensive) means to address the scaling issue, and are considered of outstanding importance within the present context.

The experience gained from developing theories or models, and from executing/analysysing experiments is available to the developers of the SYS TH codes, and are incorporated into the codes based on the feedback from users of the code. In this context, these numerical codes are the final repository of information, including the scale-related information, and have the potential capability to support scaling issues consistent with the current progress of the technology.

The set of consistent code description documents, Relap5 Developmental Team, 2001, Bestion, 1990, and Ha et al., 2011, related to three different SYS TH codes are taken as significant examples: the SYS TH codes concerned with those examples are based on complementary- and independent-experimental campaigns. The application of those codes within the technology of nuclear reactor safety implies the consideration of issues like; nodalization (or 'input deck', or set of input conditions) development criteria, and qualifications, e.g. Bonuccelli et al., 1993, code-user effect, qualification and training, Aksan et al., 1993, OECD/CSNI/NEA, 1998, and D'Auria 1998, the demonstration of code verification and validation (V & V), e.g. D'Auria & Galassi, 1998, and IAEA, 2014, accuracy quantification at qualitative- and quantitative-levels, Ambrosini et al., 1990, and Kunz et al., 2002, and the so-called Kv-scaled calculation, e.g. D'Auria & Ingegneri, 1998, and Martinez-Quiroga et al., 2014. All those issues are connected with scaling which is addressed in the present document.

The validation part of V & V process addresses the assumptions made during the development of the code. Furthermore, the Validation process allows the identification and the quantification of code errors based on comparisons with experimental data. The need for uncertainty evaluation, i.e. expecting an error when performing NPP-related calculations, also can be seen as an outcome of the Validation process. The uncertainty methods, already discussed as a background element in Fig. 1-1 (e.g. NEA/CSNI, 2006, and IAEA, 2010), also are needed when the SYS TH codes are used as support for addressing the scaling issue.

The scaling elements discussed under Category 3 are needed for the BEPU approach for safety evaluation of NPPs. This is emphasized in Fig. 1-2: scaling is relevant (at least) for the following items:

- Development of the codes.
- Performing Verification and Validation (V & V).
- Developing and qualifying input-decks or nodalizations.
- Training of code-users.
- Performing uncertainty analyses.
- Confirming/validating the final result of the targeted application.

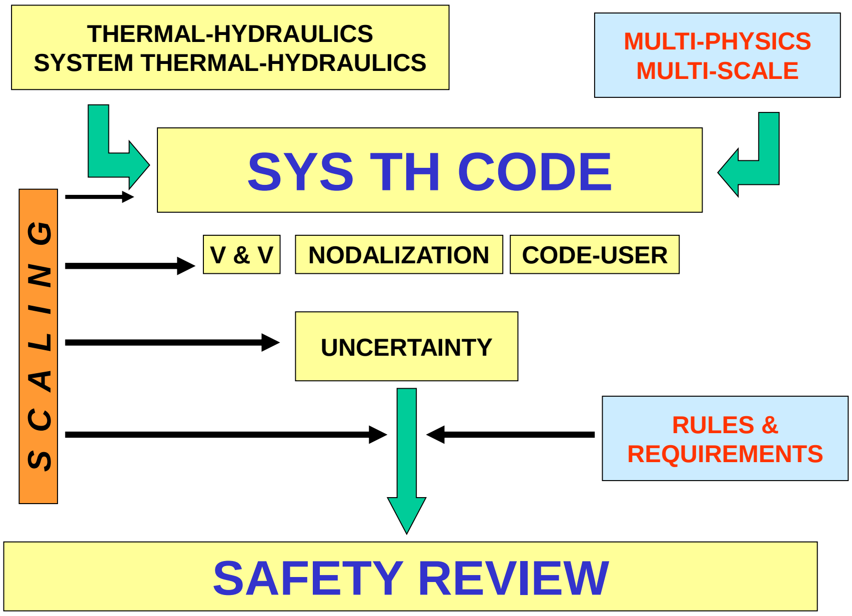


Fig. 1-2 – The role of scaling (or addressing the scaling issue) in SYS TH codes application.

Achievements from scaling activities

A wide variety of scaling activities resulted in numerous applications of scaling approaches and methods that are listed in the references provided so far; it constitutes part of the Knowledge Management for Scaling (Fig. 1-1). All those applications are valuable for creating the scaling technology and the related state-of-the-art (present report). To facilitate a common understanding, selected applications of scaling approaches are mentioned hereafter as scaling achievements. In relation to those achievements, a straightforward and understandable connection can be established between the scaling results (discussed in the relevant documents) and the ultimate target for scaling analysis (i.e. the safety demonstration for NPPs); these topics are discussed in more depth in Chapters 2 to 4. Selected examples of scaling achievements are presented here.

- NPP data: Very few accidents involving transient two-phase conditions have occurred in operating reactors so far. Even though the NPP instrumentation is poor for a suitable characterization of the transient scenario, full-scale data are available and have been used to validate scaling approaches, e.g. the TMI-2 accident, NEA, 2009: In other terms, the application of SYS TH codes to the analysis of NPP events (i.e. not only the TMI-2 accident) performed with suitable V & V procedures and successfully completed is considered a supporting demonstration of code-scaling capabilities, though related to the (typically narrow) ranges of variations of physical parameters.
- Centrifugal pump's performance: Two-phase degradation data of the centrifugal pump at different scales indicated that largest body of data at the 1/3 scale could represent the prototype pump because the degradation decreases with size. This was shown in a CSAU document (Appendix L, Boyack et al., 1989).
- Overall scaling procedure: A procedure to demonstrate the scaling capabilities for the SYS TH codes supported by scaling facts, scaling pyramid, and scaling bridges was proposed by D'Auria & Galassi, 2010.

- Flooding and CCFL: Experimental data on flooding, i.e. related to counter-current steam and liquid flow in a gravity environment at different scales, have been collected from different research projects. The downward penetration of liquid in the presence of an upward flowing vapor is strongly affected by the size (scale) of the facility, ITF or SETF, where the experiments were performed, and full scale data as reported by Glaeser & Karwat, 1993, and USNRC, 1993. The conclusions from ECC bypass studies in the 70s are directly affected by these new findings. The evolution of CCFL phenomenon largely depends upon scaling, i.e. on the dimensions of the model.

The data in the last paragraph can be seen as a demonstration, one among others, that the extrapolation of experimental data from model to prototype in SYS TH is not feasible. Parallel to this, we note that the extrapolation of calculated data alone, i.e. without the support from experiments at different scales, also is not feasible. This is discussed by D'Auria & Vigni, 1985. Therefore, the scaling achievement here is either the extrapolation of experimental data alone, i.e. data measured in similar facilities having different sizes, or the extrapolation of calculated data alone, i.e. the results from predictions related to similar systems having different sizes, are not supported by experience, and are not feasible.

The above scaling achievement, rewritten as "no extrapolation feasible or realistic (or recommended) of experimental data alone or of calculated data alone" does not preclude the possibility to extrapolate the error or the accuracy in predicting the experimental data: this is acceptable in cases where the proper conditions are fulfilled. Namely, accurate extrapolation can be used as the basis for predicting uncertainty, e.g. Martinez-Quiroga et al., 2014, and D'Auria et al., 1995, and D'Auria & Giannotti, 2000.

A historic overview of nuclear thermal-hydraulics

The NST includes accident analyses and nuclear thermal-hydraulics. Scaling constitutes a key element for the latter process. Thus, a vision of the history of nuclear thermal-hydraulics may contribute to on the understanding of scaling and related issues.

A historical perspective for nuclear reactor thermal-hydraulics is provided by D'Auria, 2012, and is summarized in Appendix A-2. The elements of scaling defined in Fig. 1-1, as well as the scaling activities connected with the application of SYS TH codes, Fig. 1-2, also are considered in the same appendix.

1.2 Objectives

The systematic consideration of the scaling elements discussed in the previous section constitutes an inherent objective for the activities undertaken by the SSG members. Therefore, the objectives for the SSG members' activity and the objectives for the present document are summarized as follows:

- Definition of the scaling issue. This includes an agreed definition for the terms 'scaling', 'scaling issue' and 'addressing the scaling issue'.
- Achieving a common understanding (or, reaching a consensus) in relation to scaling. This implies producing a common view about scaling, and also addressing the scaling controversy mentioned in the previous paragraph.
- Recommending the best practices to perform a scaling analysis. This implies considering the key objectives for scaling.
- Considering the connection between scaling and the NPP safety evaluation (the legal part referred as the licensing process). The basis for this is the application of SYS TH codes.

– Identifying the need for additional scaling activities. This is consistent with the current trends and the perspectives in system thermal-hydraulics.

1.3 Structure

The structure of the document shall correspond to the scope of, and shall allow the achievement of the objectives, all of which are defined in the previous sections. Furthermore, other than Chapters 1 and Chapter 5 which constitute the fundamentals and the outcome for the activity performed by the SSG members, three main chapters are the body for the document.

– Chapter 2 briefly describes the scaling concepts, including scaling achievements. The idea of Chapter 2 is to shed light on the scaling processes, thus covering selected scaling elements from Fig. 1-1, and selected topics from Fig. 1-2, i.e. covering all the scaling categories considered above. However, the key focus is on the scaling requirements
– Chapter 3 provides a systematic overview of the scaling elements focusing on existing/applied techniques and approaches. Thus, the main focus is on Category 1 and Category 3. The scaling models (analytical or numerical) and the experimental programs, i.e. the facilities and tests including the scaling rationale for both, are considered in this chapter.
– Chapter 4 is concerned with, a comprehensive vision on the SYS TH codes as tools to perform scoping scaling analysis, and deal with code uncertainty in predicting phenomena. A scaling road-map is proposed as a way to address the needs coming from the BEPU process.

Summing up, the scaling elements constitute the subject of Chapters 2, 3, and 4. The application of assessment of code scaling capability constitutes the subject of Chapter 4, and all remaining elements are considered in Chapters 2 and 3.

Furthermore, the present document includes also the Foreword, Abstract, Executive Summary, Glossary, List of Acronyms and References, and five Appendices. The Appendices deal with: A-1) the planning document (so called CAPS) at the basis of the activity for issuing the current S-SOAR; A-2) historical remarks for nuclear thermal-hydraulics also showing the role of scaling; A-3) lists and characteristics of experimental facilities designed on the basis of scaling and suitable for addressing the scaling issue; A-4) outline of the processes of system thermal-hydraulic code development, verification and validation; A-5) Extended Executive Summary.

2. SCALING ISSUES

2.0 Introduction

As discussed in the First Chapter, scaling is a process of demonstrating the applicability of the scaled parameters to the reactor's conditions. The scaling issues refer to the complexity of the scaling process, and its associated aspects. In this chapter, the complexity of the scaling process and its related subjects are reviewed.

The concepts and techniques of scaling have been widely used by scientists and technologists for centuries. In the past few decades, scaling technology has evolved from primitive algebraic approaches (e.g. the Buckingham Pi theorem) to complex mathematics/physics formulations. In this evolution, many obstacles and limitations of scaling were uncovered. With the advances in nuclear-power-plant technology, the plant's design and operation rely more on the computer code safety analyses. Nowadays, the assessment of reactor safety under design-base accidents or plant transients is accomplished through the use of computer codes that have been created for this purpose. Figure 1-2 demonstrates the role of scaling in this process. These codes predict the Figure of Merit (FOM) such as peak clad temperature (PCT), and containment pressure, and the safety margins are ascertained by comparing them with the acceptance criteria established by the regulators. These computer codes are collection of balance equations for two-phase flow with constitutive relationships, coupled to neutronics. However, confidence in the safety margin of the figure of merit requires a statement of uncertainty in the prediction. The uncertainty in FOM prediction arises from uncertainties in the constitutive relationships in representing the phenomenon at the scale of the NPP and thermal-hydraulics conditions, and from numerical approximations [CSAU, OECD/CSNI reports on UMS and BEMUSE and IAEA SRS 52].

Before computer codes can be used for safety evaluation, they must be assessed for their applicability for the intended plant and the transient. This is done through a code-validation process. While applicability can be determined by reviewing code's documentation, the validation is established by comparing the results of the code's prediction for separate- and for integral-effect tests with the data. These tests should scale the phenomenon expected in the plant. As scaled tests are essential for reactor safety, especially for obtaining the best estimate with the uncertainty-estimate approach (BEPU), the scaling methods have to be designed. These scaling approaches in ideal case are applied before the tests are designed and run. In practice, the tests have already been run, and extensive scaling assessments and evaluations of distortion have to be done after the fact. In addition, as the range of thermal-hydraulic conditions change from plants and transients, it may not be possible to design tests for all conditions and geometries, and so a scaling evaluation will be needed for each application, Wulff & Rohatgi, 1999.

Due to the reality that a full-size reactor experiment is not achievable, and the codes only can be validated from the data obtained in scaled experiments. Scale distortions affect the estimate of uncertainty in FOM, and the safety margins. Therefore, the requirement for and the evaluation of scaling in, regulatory process is necessary to ensure the safety decision. On the other hand, to overcome the scaling distortions and limitations, new scaling techniques were developed and applied in scaled experiments. Nevertheless, due to the lack of actual data from full-sized plants data, it is difficult to verify the scaling laws. Hence, scaling has remained a significant source of uncertainty.

In the beginning of this chapter, a picture of the scaling world to depict the subjects of scaling from several perspectives is presented. The purpose of the picture is to give readers the scope of this SOAR subject. In the picture, a summary of activities that involve scaling is introduced, including activities that prompted the development of the milestone scaling technique and applications. The second topic embedded in the picture is an introduction to the relationship between thermal-hydraulic scaling and nuclear reactor safety. Finally, the major achievements that scientists and engineers have accomplished in scaling are briefly documented.

Following the scaling world, a serious topic in scaling will be reviewed – scaling distortion. It is well known that distortion is the center of all scaling issues. Its origin is described here, using some well-known examples. The distortions could arise due to factors like assumptions and simplifications in the analytical methods, limitations in the constructing and operating experimental setups, and scalability issues embedded in the computer codes. It is well recognized that distortion is inevitable in scaling a complex system like the nuclear-reactor system. Therefore, an acceptable way to quantify the distortion must be devised to guarantee the quality of the data for safety analysis. The physical meaning of the parameter used in quantifying distortion and its relationship to the figure-of-merits in the transient must be explained. In a thermal-hydraulic transient, the plant's behaviours usually are considered for different phases, according to key events. Scaling laws usually were derived from the physics describing the thermal hydraulic processes that dominate in a particular phase. This means that the geometry and operating conditions derived in one phase could introduce distortion in another phase wherein the process becomes less dominant. Therefore, the impact of distortion on the figure of merit could propagate from one phase into the next phase. If this propagation during the transient is not considered, it may remain unaccounted for.

Another element in scaling issues is scaling the complex thermal-hydraulic phenomena in an experiment. Most of these phenomena are stochastic and local, therefore, are difficult to describe with standard field-equations. They usually affect the operation of the emergency core cooling (ECC) system, and cannot be neglected in the scaling process. Due to their complex and chaotic nature, empirical correlations are normally used to derive the scaling laws. This poses a great challenge to scalability since the correlations usually have been developed in scaled environment. To improve their resolution, separate effect tests are usually needed for these phenomena. The decision of choosing an integral effect test (IET) or a separate effect test (SET), or using both, is another challenge in scaling. We are trying to introduce the basics of design and choice between IET and SET. The scaling bases of existing IET facilities and SET facilities are reviewed and summarized.

To ensure nuclear-reactor safety, regulatory agencies are tasked with the responsibility of reviewing nuclear-reactor design and operation. Reactor vendors apply for a license by submitting the safety analysis of design and operation. The safety review usually starts with the tools used in the safety analysis. Scaling plays an important role in developing and assessing the tools. To illustrate the relationship between scaling and the regulatory requirements, the authors used two regulatory processes currently used by NRC, as an example – viz., the Evaluation Model Development and Assessment Procedure (EMDAP), USNRC, 2005 and the Code Scaling, Applicability and Uncertainty (CSAU) methodology, USNRC, 1989. The requirements in these two procedures are the technical bases of addressing scaling issues. In Section 2.4.2, the relationship between scaling and regulatory requirements is elaborated.

The following topics are covered in this chapter. The main goal is to usher the readers into a detailed review of the scaling issues to be documented in subsequent chapters.

– A picture of the scaling universe

Scaling activities (targets)
Scaling needs for nuclear-reactor safety
Achievements from scaling studies

– Scaling distortion

Origins and examples
Deficiencies in scaling methods
Deficiencies in system codes and CFD
Quantification (treatment) of scaling distortions
Propagation of scaling distortions

– Scaling in complex phenomena and test design

TPCF – two phase critical flow
CCFL – counter current flow limitation
Entrainment and de-entrainment
Reflood
ITF and SETF design and operation
Fuel-rod ballooning
Special components

– Address scaling issues

Evaluation of model development and assessment
Requirements of the CSAU

2.1 An overview of the scaling universe

Scaling is an important issue in reactor safety due impossibility of carrying out reactor-scale tests under reactor conditions. In the following sections, this topic is explored for its role in NPP safety evaluation, and its application to the design and the analysis of separate-effect tests and integral-effect tests.

2.1.1 Scaling activities

Nuclear reactors are combination of various components with different geometry and orientation, friction losses, and heat transfer. Also included are many safety systems, such as coolant injection, safety valves, or control rods that are designed to safely shut down the reactor system under any accident scenario. It is important that these safety systems be assessed for their performance under accident conditions. However, it is not possible to run tests at the nuclear-power plant. Therefore, the strategy is to use a combination of computer codes and tests at different scales. These tests include separate-effects tests representing a single phenomenon, and integral-effects tests representing interaction of different components and phenomena under various transient situations. To assure the relevance of these tests, they should represent the NPP under the correct thermal-hydraulic conditions expected during postulated accident conditions.

Test facilities for single phenomenon, or separate effect tests are needed for three possible applications for the codes. First, the codes need constitutive relationships or correlations to model flow regimes or interfacial areas and shapes, correlations for transfer of mass, momentum, or energy at the interface between phases, or between fluid and solids. The second need is for validation of the code. This is done by modelling the tests with the code, and comparing the predicted values and the measured ones. If the predictions are within the experimental error margins for the data, we can conclude that the code can model the phenomenon very well. A review of the documentation of the constitutive relationships and their basis, and the validation studies, together establish the code's applicability, that is, the code is capable of modelling the phenomena.

In addition, reactor safety is assessed by codes that integrate various phenomena and components, and predict figure-of-merit; it is essential to provide an estimate of uncertainty in the predicted figure of merit since the codes integrate different phenomena, they also can be used to combine and propagate different uncertainties in the models. The uncertainty in an individual phenomenon is estimated by using separate effect tests. These tests should the scale the phenomena in the plant. If they do not scale well, there is further need for estimating scale distortion. There are other models in the code that are not based on first principles but on correlations of performance, such as pump, critical flow, and the CCFL. These correlations also should come from components that scale similar components in the plant. This brief description shows the need for scaling.

Beside separate effects and component tests, there are integral effects tests. This group of tests is needed to understand system level behaviour and the interaction of various components. These test facilities also should be scaled to represent as many transients as possible in the NPP. There are two other benefits of integral effect tests. First, the data can be used to validate the system computer codes, and its ability to combine different models/correlations. Second, the error in measured data in some cases may be used as the target precision needed for phenomena predicted by codes.

Computer codes are used for simulating many types of transients, such as LBLOCA, SBLOCA, station blackout, instabilities, and ATWS. In each of these transients, the reactor components experience different thermal-hydraulic conditions, such as flow rate, pressure, sub-cooling, and void fraction.

The integral test facilities are designed for one phase (intended here as phenomenological window like blowdown, refill and reflood during LBLOCA) or one class of transients, and are used to check for other phases. These phases are characterized by few dominant phenomena that need to be preserved in the test facilities. There will always be scaling compromises to represent each phase reasonably well. There always will be scaling distortions. Further, as these facilities are expensive, they may be used for other transient experiments beyond their original purposes. For example, LOFT originally was designed for a LBLOCA study. However, later it was used for SBLOCA and some plant transients. Separate effects tests generally are for a single phenomenon. They are used to assess a code's ability to model a single phenomenon under expected reactor conditions.

Scaling is assessed first by developing non-dimensional groups-based on facility dimensions, and fluid conditions, and then by comparing the values of these groups for two facilities. Under single-phase flow, the number of scaling groups is limited so it is easier to assess scaling. However, in two-phase flow there are many degrees of freedom, and, therefore, there are many non-dimensional groups, and invariably, not all groups can be matched. However, scaling analyses also provide a method of ranking different phenomena in terms of their importance to the figure-of-merit. This is done by evaluating different non-dimensional groups, and their values represent the significance of the underlying phenomenon. It is important in designing a facility, that it scales the prototype for important phenomena.

In the early days of scaling, two approaches were applied. One global approach was based on identifying different dimensions and fluid parameters, and developing non-dimensional groups using the Buckingham Pi theorem. While this approach provided groups, but these groups may not have any physical meaning. The other approach was more a local scaling. Here, the differential equations, representing conservation of mass, momentum, and energy, are non-dimensionalized with reference values that are order-of-magnitude of the variables, both independent and dependent in the equations. This leads to a set of local non-dimensional groups, such as Reynolds number, Froude number, etc. These groups do have physical meaning. The objective here is that non-dimensional groups with the same values, similar initial- and boundary-conditions will lead to the same solution in non-dimensional space. The flows will be similar. This approach works only for simple flows, and 'simple flows' either do not exist, or are of low interest in reactor safety. Furthermore, the differential equations may include empirical constitutive equations that usually are only valid within restricted ranges of parameters, and consequently, the non-dimensional groups are constructed based on empiricism. Also, not all sets of differential equations have been numerically solved and, in principle, the set of equations may be elliptic (i.e. no real solution) at least in some regions. So, we have the paradox that non-solvable equations are used to derive non- dimensional groups. This approach has been applied with systems where length scales change and non-dimensional groups change at different locations. There could be very many non-dimensional groups when these equations are integrated, Kocamustfaogullari & Ishii, 1987. A simpler example of this approach is its application to boundary-layer region (Reynolds number) and in determining the importance (Grashof number and Raleigh number) of buoyancy in natural convection.

In supporting nuclear reactors, test facilities have been designed to simulate system behaviour and separate effects tests have been established for a single phenomenon. The scaling approach is multi-stage hierarchal approach, e.g. Zuber, 1998, Zuber et al., 2007, Wulff, 2005 and Catton, 2005. The transients are divided into phases (time periods) based on the possible dominant phenomenon. For each phase, a control volume is defined as a system. It will encompass the interaction of processes and boundary conditions. The integral forms of conservation equations are written. The equations will have a storage term for system level variables (quantity) and processes that affect this variable. The reference quantities are estimated either at the beginning of the phase or at its end. The non-dimensional groups or coefficients of process terms provide the relative effect of the processes and, therefore, their ranking. The high-rank processes or components can be scaled further in same way. This step may identify important processes at a lower level that have the most impact. This approach allows for designing test facilities that capture the most important phenomena. However, one underlying requirement is that the geometry should not become so small so as to change the nature of the flow regimes due to distortion. The acceptability of scaling distortion is surveyed with more detail in Section 2.2.4.

The computer codes have correlations and empirical criteria along with balance equations. These correlations relate to mass, momentum, and energy transfer at the interfaces between, gas, liquid, and solids. Many transfer coefficients either are derived from analyses or from test data that were plotted in non-dimensional space, such as heat transfer coefficients. The correlations are developed by fitting a curve to the data. It is noted that these coefficients do depend on the shape and size of the geometry of the interface. There are other sets of empirical relationships that relate to flow regime transitions. These relationships also are empirical. The flow-regime maps are affected by the flow rate, orientation, pressure, void fraction, and the cross section of the flow path. Predicting the correct flow regime is important as the interfacial area and the transfer coefficients will depend on the flow regimes.

The safety review of a proposed reactor design requires submission of system analyses for design-basis accidents and transients to show the performance of safety system. As there are no full- scale tests, the only assessment for the safety system is through computer codes. However, before the results are accepted, it is necessary to show that the code is applicable, and that there is a statement of uncertainty in the predictions of safety parameters. The applicability of the codes is determined by reviewing the documentation to see if the required models based on PIRT are in the code, and then by modelling relevant separate-effects tests with the codes. The predictions are compared with the data, and if the predictions fall within the instrument's uncertainty, the code has accurate models. In cases where there are large differences between the predictions and the data, the differences will affect the estimates of uncertainty. It is important that the tests are used for validating the code and they are properly scaled to the plant, Wulff & Rohatgi, 1999. Once it is established that the code is applicable, a statement of uncertainty in the safety parameters is needed. This is estimated through a systematic incorporation of the uncertainties and the distribution of uncertainties for each important correlation, boundary condition, and the initial conditions in the code. There are systematic approaches for combining all the individual uncertainties in the prediction of safety parameters. It is controversial as to whether the scaling distortion belongs to uncertainties. It generally is acceptable in the nuclear industry that the overall uncertainty of a predicted safety-parameter includes the scaling distortion because the scaled tests are used to estimate uncertainties in the individual phenomenon that are aggregated with instrument uncertainty, and numerical uncertainty for computing overall uncertainty. Safety requirements are that the predicted safety parameter meets the specified criteria with 95% confidence.

2.1.2 Scaling needs for nuclear reactor safety

USNRC Regulatory Guide (RG) 1.203, USNRC, 2005, offers a detail discussion for procedures and acceptance criteria for nuclear reactors. This guide describes design basis accidents and transients that are supposed to cover the most probable events. However, it is not possible to obtain data for these transients from the plants to assess the effectiveness of the safety systems. This assessment is done by simulating these transients with computer codes designed for this purpose. RG 1.203 describes the requirements for codes and methods as described in Section 2.4.1 of this chapter.

The computer codes are combination of balance equations, and closure- or constitutive-relationships. The formulation is an approximate representation of actual conditions. There are approximations in selecting the set of balance equations, such as two-fluid- or mixture-approaches, nodalization, and the type of constitutive relationships that normally are derived from stand-alone tests. These tests are at lower scale than the plant, and sometimes at different thermal-hydraulic conditions. The computer codes are the tools for combining all the constitutive relationships and applying them for full plant conditions.

There are three different needs for the experiments. First is for tests that are used or have been used to develop constitutive relationships. The second need is for code validation wherein important phenomena are simulated at scaled facilities. The third need is for uncertainty analyses. All these sets of tests should be a scale representation of the plant as far as possible.

The first step in applying a code to simulate a plant transient is to determine the code's applicability. This is done by reviewing the code's documentation and the tests behind different constitutive relationships. If the tests are full scale, or a scaled version of the plants, the constitutive relationships derived from the data will be applicable to the plant. In addition, code validation is performed where the code is used to simulate separate effects (SET), and the integral effects tests (IETs), and the predictions are compared with the measured values. The validation work is the second need of scaling. If the tests are reasonable scaled version of the plants, and code predictions are within the measurement uncertainty, the code is considered applicable for the simulating the plant.

To have confidence in the performance of the safety system during design-base accidents, an estimate of uncertainty in the prediction of figure of merit, such as peak clad temperature, is needed. As the computer code integrates different phenomena over time, it also is a tool for combining and propagating uncertainties in different constitutive relationships. The uncertainty in a constitutive relationship is estimated by comparing the prediction of a representative parameter for the phenomenon with the data from the scaled tests. The uncertainty in the representative parameter will have contributions from measurement uncertainties and scale distortions. A mean value and distribution of uncertainty in the parameter is obtained from modelling single phenomenon separate effect tests (SETs). Therefore, scaling studies of the tests used for uncertainty analyses are the third need for the assessment of safety.

2.1.3 Achievements from scaling studies

Scaling has been an established subject in nuclear-reactor technology at the time of writing the current S-SOAR, as already discussed in previous sections: namely, the subject started from the need to scale-up the capabilities of thermal-hydraulic models has been clear since the 50s; later on (in subsequent decades), the same need was extended to computational tools. Therefore, one could expect many significant findings or achievements to be available from scaling studies.

It is not the purpose for the S-SOAR to become an encyclopedia of scaling studies, neither the objective for the present section to collect in a comprehensive, systematic way the achievements from the existing scaling studies available from the literature. Rather, the idea is to follow-up on the concept of the scaling strategy depicted in Fig. 1-1 of the Introduction, and to focus on the top level of strategy (scaling achievements), by providing significant examples. Then, the objective for the present section is to demonstrate what was written in the first sentence of the previous paragraph, i.e. that scaling is an established subject, and not just an issue, at the time of writing of the S-SOAR.

Examples of scaling achievements are provided hereafter; more details are also discussed in other chapters of the document. These examples are Sections 2.1.3.1 to 2.1.3.6:

- Flashing, flooding, and the counter current flow limitation in a downcomer of a PWR reactor-pressure vessel during large break loss-of-coolant accident;
- Wall evaporation, flooding, and countercurrent flow limitation in downcomer of steam generator secondary side, during accident recovery conditions;
- Influence of the number of tubes on the natural circulation in primary side of the U-Tubes steam generator;
- The concept of scaling of accuracy (also known as Accuracy Extrapolation) in the uncertainty method based on the accuracy of extrapolation is an active research area;
- The concepts of a Scaling Pyramid;
- The simulation in integral test facilities of nuclear-fuel rods by electrically heated rods with and without a gap.

2.1.3.1 CCFL in RPV downcomer originating from flashing in the lower plenum

The focused scaling issue here is the design of the RPV downcomer and the experimental demonstration of the capacity of ECCS during a large-break LOCA. The downcomer's width in a suitable ITF is the target parameter. The aim of the experimental design is (among all considerations) the correct simulation of the penetration of ECCS liquid from the intact loop's cold legs to the lower plenum, and to the core. This phenomenon occurs during the blowdown period of a LBLOCA, following the injection of ECCS water into the non-broken loops of a PWR. Resolving this issue is essential for designing integral test facilities (ITF) which are used to prove the capabilities of computational tools.

A pioneering investigation of the subject phenomenon was performed in the Semiscale facility installed at Idaho National Laboratory near the end of the 60s, Batt & Berta, 1978. During the simulated LOCA conditions, the flashing of liquid from the lower plenum prevented the penetration of the ECCS-injected water. The coolant injected by the accumulators during the blowdown period was found experimentally to be almost completely diverted toward the rupture, causing the bypass of the core. The resulting phenomenon was called the 'ECCS-core-bypass.' The ECCS core-bypass is affected by many factors, including the depressurization rate, the two-phase critical flow at the break, the pressure drops inside the vessel, and the heat transfer from the RPV walls. The last phenomenon also is known as the 'hot-wall-delay'. The design of ECCS and its ability to cool the core became of concern in reference to the inability to cool the core during the fast depressurization of the LBLOCA observed in the experiment. Subsequent in depth analyses with continuous experimental support continued until the availability of the full-scale UPTF experiments in Germany, Damerell & Simons, 1992. The issue then was recognized to be an ITF-SETF design-scaling issue.

Scaling Achievement

The radial size of the RPV downcomer strongly affects the penetration of liquid injected by the ECCS, mainly from the accumulators, and the resultant peak cladding-temperature during the LBLOCA blowdown phase. This was confirmed when all key LOCA parameters were kept constant: noticeably, the depressurization rate, the ECCS design features including the actuation pressure, lower plenum volume, the RPV wall temperature, and height of the downcomer region. Consequently, scaling distortions are expected in an ITF when power-to-volume scaling is adopted, as well as the full height of the RPV. In this scaling approach, the downcomer's height and the gap width are fixed by the flow-area/volume ratio. The distortion is described in more detail in Section 2.2.1.

2.1.3.2 CCFL in SG downcomer originating from the evaporation at the wall

An important scaling issue is the experimental simulation of the effectiveness of cold water injected at the level of nozzle of the auxiliary feed-water (AFW) in the secondary side of the SG. The liquid is injected at low pressure and nearly-ambient temperature inside an empty, depressurized SG with outer massive walls at a temperature close to the nominal operational temperature. The effectiveness is determined by the capability of the injected liquid flow to cool the bottom part of the U-tubes. The issue is related to the design of accident-management procedures (AMP) that the plant is under prolonged (typically outside the envelope of the design basis accident) station blackout situations, and when the primary circuit is at a pressure close to the nominal operational value. In that situation, coolant is available at the NPP site with pressure around 0.5 MPa, which is deemed possible to feed the depressurized secondary side of the SG.

The issue was brought to the attention of the scientific community in the analyses, Annunziato et al., 1993, of the LOBI experiment, BT-17, performed in the 80s at the European Commission establishment of Ispra (Italy). Basically, the AFW cold liquid was injected at about 10 m elevation relative to the bottom of the U-tubes, with a flow-rate consistent with the accident's progression. It took a few minutes during the experiment for the liquid to reach the bottom of the U-tubes and to effectively cool the primary circuit in order to restore the natural circulation, and to quench the core. The reason for the few-minute delay was found to be evaporation at the SG's hot walls. The evaporation on the wall created a flooding-CCFL condition, preventing the liquid from reaching the bottom of the SG on the secondary side. The challenge then was the lack of capability in the code to simulate the few- minutes delay. Instead the code predicted the formation of level in the bottom of the SG soon after the injection started. Consequently, the injected AFW liquid in the bottom immediately restored the natural circulation in the primary loop and also restored the cooling of the core from a degraded situation. In the experiment, however, the core was not quenched, and clad temperature went up to the threshold value, and tripped the electrical power to prevent fuel damage.

In the example above, the issue appeared to be associated with the capabilities of the adopted computational tools, rather than the scaling. However, scaling plays a role in determining the width of the ITF SG downcomer and it is not known if the experimental results faithfully reflected the phenomena in the prototype reactor. One may note that depressurization played an important role in vaporization in the example in Section 2.1.3.1, while the vaporization of falling liquid due to a hot wall is the key role in the case of the steam generator. It also should be noted that the hot-wall effect also contributes to ECC bypass. Furthermore, AMP was designed in existing NPPs based on the injection of cold liquid at high elevation in a depressurized SG. Currently, the experimental data with correct scaling, which is suitable for code validation for CCFL phenomenon, is not available.

Scaling Achievement

This scaling case is a concern, rather than an achievement. In this case, SYS TH code simulation results are used, e.g. Annunziato et al., 1993, for designing an experiment in which the AFW liquid is injected into an empty zone of SG downcomer with high wall-temperature. It is urgent (important) to demonstrate the code' capabilities in simulating the interaction between the cold falling liquid and the hot walls of the thin

SG downcomer test facility. The concern here is that AM procedures developed considering this experiment already are part of existing NPPs' safety procedures, and as yet, there is no comprehensive demonstration of the code's capabilities in simulating the subject phenomena.

2.1.3.3 NC performance and scaling of the number of U-tubes in the SG

The scaling issue of concern here is the minimum reasonable number of U-tubes per each SG needed to correctly simulate the conditions of natural circulation in the primary circuit of a PWR during a variety of small break LOCA events. The origin of the issue involves three aspects: a) the relatively small value of coolant velocity in U-tubes, which may cause adverse pressure relationship (i.e. flow reversal from outlet plenum to inlet plenum) between the inlet- and outlet-plenum of the SG under the small driving force due to gravity; b) the presence of different-length parallel flow paths between the inlet- and the outlet-plenum of the SG: more than 4000 tubes with a top elevation ranging between 8 m and 12 m are installed in a typical NPP prototype SG; c) the three-dimensional flow distribution inside the SG inlet- and outlet-plenum: the plenum geometry and the upstream conditions (e.g. in HL) may affect the liquid and steam flow distribution to the tubes.

From the operation of several ITFs, the following situation was encountered in both single-phase (liquid) and two-phase conditions, e.g. D'Auria et al., 1992, D'Auria et al., 1991, and Umminger, 2012:

- ⇨ Flow reversal occurred in several U-tubes, causing the circulation of flow from the outlet plenum to the inlet plenum of the SG (inlet and outlet are referred to the nominal operation of the RCS of the PWR), while most U-tubes work under the expected nominal flow direction.

The following are the consequences of the flow reversal: A) The hydraulic resistance of the RCS during NC conditions increases; B) a suitable number of U-tubes in the ITF are needed to simulate the SG's performance. The former issue is related to code validation (thus, imposing the need for several parallel U-tubes in the nodalization), and is indirectly related to scaling, and the latter (B) is directly related to scaling.

Scaling Achievement

The SG part of an ITF should justify a reasonable number of U-tubes of the same height as in the prototype so to simulate NC conditions. The scaling achievement includes the availability of experimental information of flow reversal, and the discovery that several U-tubes are needed to simulate the NC performance of a PWR RCS. The minimum number of U-tubes needed for a correct code prediction could be estimated to be close to ten in each SG. This estimated minimum number of U-Tubes may also be used to determine the minimum size of the SG for a suitable ITF. Furthermore, the design of the ITF should also include instrumentations that allow the characterization of the individual tubes (flow direction and pressure differences).

2.1.3.4 Scaling of Accuracy and the UMAE procedure

Scaling is not only a key element of CSAU, USNRC, 1989, and it is also a major step of the pioneering uncertainty methodology proposed by US NRC. To realize the connection between scaling and uncertainty, PIRT is proposed for inclusion in the scaling step. However, some drawbacks of PIRT were pointed out by N. Zuber, (Zuber, 2010), the lead author for CSAU, about 20 years after its publication.

UMAE procedure (D'Auria et al., 1995, see also Section 4.4.2) makes the scaling requirements of CSAU workable. CSAU requires that a code should be qualified against scaling. A procedure in UMAE addresses this requirement. For a designated phenomenon that is characterized by a set of parameters, accuracy is evaluated by using the experimental values, and the code-calculated ones for the parameters. When some pre-defined conditions are satisfied, particularly the demonstration that the accuracy (or the unavoidable code-calculation-error) is independent of scaling, the accuracy itself can be extrapolated to determine the uncertainty in NPP calculation for the subject phenomenon and its representative parameters.

The UMAE procedure has been automated (i.e. the error is calculated by a specific software in which both the available experimental data and the calculated data are used) within the CIAU procedure, D'Auria et al., 2000.

Scaling Achievement

The achievement here is the connection between scaling and uncertainty. The connection is to fulfill the requirement of CSAU code scalability (published 1990) that is made workable by UMAE (1985) and CIAU (2000). Furthermore, the second achievement is the possibility of extrapolating the accuracy, provided that a suitable number of experiments at different scales are available, and the qualification processes of UMAE and of CIAU are met.

2.1.3.5 The concept of the Scaling Pyramid

The knowledge of the NPP's true performance in transient conditions is the ultimate goal of any scaling analysis. In the cases when the true performance of the NPP can be obtained from measured plant data, the scientific interest in scaling naturally will decline. Imagine if we designate the true NPP performance as the tip of a pyramid describing the technology, and current knowledge as the basis of the pyramid. Several lines of connection can be established between them (D'Auria & Galassi, 2010). D'Auria & Galassi, 2010, have shown that conduction heat transfer in NPP structures, even in transient conditions, may be estimated without a specific scaling study once the computational tool is qualified to be scale independent. It is true for other thermal-hydraulic processes in the accident analysis. On the other hand, a series of counterpart tests and similar tests (Section 3.3.1) may provide an understanding of NPP performance without computations tools if sufficient large-scale ITFs are available.

Scaling Achievement

Scaling analyses can be used to connect findings and established knowledge in thermal-hydraulics with the expected NPP transient performance under a variety of conditions. The scaling achievement is that some technologies related to scaling are available to utilize the existing comprehensive knowledge to address, or even possibly to close the scaling issues. Scaling-independent computational tools (system thermal-hydraulic codes and nodalizations) are needed in this context (Section 4.4).

2.1.3.6 The simulation of nuclear fuel in the ITF

The experimental simulation of the reactor core and individual fuel rod within assigned boundary conditions (see below) necessitates the design of electrically-heated rods that are characterized by their outer geometry (i.e. diameter and length) and linear power (i.e. q', w/m) equivalent to nuclear rods. It is sufficient for full-scale simulation of nuclear-fuel performance in an accident scenario, despite a substantially lower amount of stored thermal energy in the electrical rods due to the fabricated materials. However, initiatives were taken to match the specific thermal capacity with that of the actual nuclear fuel. For instance, CHF conditions are created in electrically heat rods, and algorithms have been established to calculate expected CHF conditions in nuclear fuel. Phenomena, such as early core quenching or rewetting during blow-down (e.g. caused by non-ECCS fluid entering the core) are strongly affected by the stored thermal power under nominal operating conditions. However, it can be experimentally reproduced with current technology, e.g. by controlling electrical power as a function of time during blow-down. However, this has not been realized for the reason of high cost of experiments in the scientific community but it should not be considered a scaling deficiency.

Scaling Achievement

Established design and manufacturing capabilities exist for simulating full-scale nuclear fuel when intact fuel geometry or un-deformed fuel rods are desired. The conclusion may not be valid in the scenarios, like clad ballooning and transient cracking of UO2 pellets due to either a high burn-up rate or a steep transient. Therefore, the scaling achievements in this area are the established design, and manufacturing capabilities; the scaling deficiencies should be in the acceptable range of the application.

2.1.3.7 Summary of achievements

A wide variety of scaling research activities have been completed during the past half-century. In some cases, scaling was not the main concern of the investigation, but the information obtained can be beneficial in addressing scaling issues. These facts should guide the current S-SOAR when reading the available documents. Instead of proposing new scaling analyses or approaches, the focus of attention should be in evaluating existing findings and the connection of these findings to resolve the scaling issues or to prove them non-existent.

A lesson learned, from reviewing ECC bypass phenomenon in different small facilities, is that there can be change in flow regimes at higher sizes that cannot be predicted by extrapolation or scaling. Scaling analysis only is effective when the underlying physics remains the same.

2.2 Scaling distortion

2.2.1 Origins and examples

Scaling distortion refers to any discrepancy between the scaled parameters and the referenced plant parameters. In a perfectly scaled experiment, all the scaled parameters are equally reproduced at the intended scaled time. In other words, the scaled thermal hydraulic parameters of interest measured in the perfectly scaled experiment are equal to those of the referenced plant, and the phenomena and events of interest taking place at the same scaled time. There are many factors to prevent these results from occurring. An obvious reason lies in the initial- and boundary-conditions of the domain of interest. The physical dimensions of a scaled domain are part of the boundary conditions that affect the fluid's behaviour despite of the fact that the physics laws apply equally in both the prototype and the model. In a nuclear-reactor system, the coolant travels through complex flow paths and is subjected to complicated energy-transfer processes. A scaling method aimed at shrinking flow geometry easily could affect the fluids' behaviours and energy state, locally or globally, in a nonlinear way. As the fluid behaviour differs, distortion occurs. As the magnitude of the distortion increases, the behaviour of the fluid sought for in the model greatly deviates from that in the original prototype. In this situation, the model could no longer represent the physics anticipated in the scaling design. In this report we refer this situation as "scaling limitation".

A simple example can illustrate the concept of scaling distortion. Consider a problem in draining a fully filled water tank that has a leakage hole in the bottom. If we make a scaled model by proportionally reducing the tank's dimensions at a constant ratio, namely lr, then the area of the leakage hole is reduced as lr2 accordingly. The height of the water above the leakage hole in the model is linearly scaled down accordingly. Since draining is driven by the gravity force, which depends on the water's height, the velocity of the movement of the liquid at the exit will be scaled down at the ratio of the square root of the scaling ratio, lr1/2, according to potential flow theory. Therefore, the flow rate will be scaled as lr5/2 instead of the cubic power of lr. With the water volume in the model being the cubic power of the scaling ratio (smaller) and the leakage flow rate lr5/2 (smaller), the time it takes to drain the entire model tank will be different than that of the prototype. Therefore, distortion in the time occurs. To preserve the draining time, we can adjust the leakage whole area as lr 5/2 instead of lr2 to compensate for the difference in velocity scaling. Then, the ratio of the rate of leakage flow will be equal to the cubic power-scaled ratio, which is the volume ratio. Although we can preserve the time ratio this way, the draining velocity will not be linear. This simple example of draining tells us that it is hard to design a scaled model to meet all the performance criteria, particularly in a complex thermal hydraulic system like a nuclear reactor. Hence, the scaling technique chosen depends on the experiment's objectives.

Another well-known scaling distortion occurred in the Semiscale facility in 1970s. In a PWR emergency core coolant (ECC) delivery experiment, the travel time of ECC to reach the PWR lower plenum delayed significantly, Batt & Berta, 1978. The delay seemingly was due to the heat up of liquid in the downcomer region through the hot wall (core shroud), which increased the upward velocity of the gas, and held up the downward ECC. The Semiscale was designed such that core was half of PWR active core height but the coolant volume was scaled according to the volume ratio. Therefore, the flow was close to one-dimensional. According to the scaling data, the length-to-diameter (L/D) ratio of Semiscale is much higher than its counterpart facility LOFT, i.e. 24.1 versus 4.53. This means that the diameter of Semiscale is much smaller. The ratio of the heat structure's surface area to the volume ratio is derived to be proportional to the inverse of its diameter (1/D). Therefore, the surface area- to-volume ratio in Semiscale is much higher since its L/D ratio is higher with L being the same in both facilities. The hot- wall delay of ECC in Semiscale was measured as 10 seconds, and in LOFT it was 0.5 to 1.0 seconds. In a typical PWR, the L/D ratio is about 3.0, which is slightly less than the LOFT's value. Thus, the ECC hot- wall delay in LOFT was not considered significant. In this example, the method of volume scaling introduced a significant distortion in the ECC's delivery time. In addition, the large surface- to- volume ratio in Semiscale led to a large heat loss to the environment. The small cross-sectional area required an external pipe as a downcomer that was much different in shape from that of the annulus for the downcomer. This shape also affected the ECCS's flow.

Scaling distortion was inevitable, as was further demonstrated at the integral test facility known as the Purdue University Multi-Dimensional Integral Test Assembly (PUMA), Ishii et al., 1996, and Ishii et al., 1998. Distortions normally are encountered for two major reasons: Difficulty in matching the local scaling criteria, and lack of understanding of a local phenomenon itself. Therefore, directly extrapolating the local experimental data to the prototypic conditions often is quite difficult or impossible. For example, in addition to the single-phase scaling requirements, the geometric similarity requirements also must be met. With these requirements, the effects of each term in the conservation equations, as expected, are preserved in the model and prototype without any distortions. If any one of these requirements is not satisfied, then some of the processes in the model and prototype will be distorted.

For example, among the similarity requirements in the PUMA experiment, the friction similarity was difficult to satisfy individually for each component, except for components with sub-channel geometry. Also, the ratio of conduction depth and the hydraulic-diameter ratio should satisfy certain criteria. They are important mainly at the major heat-transfer components where these conditions easily can be satisfied. However, satisfying all these criteria over the entire loop in the same time may be difficult and may lead to an overall scale-distortion of structural-heat losses.

Another example of distortion in PUMA was related to the void fraction in two-phase flow. For the chimney section in the facility, it was necessary to select a (d)chim that properly simulated the two-phase flow regimes. A smaller hydraulic diameter was chosen to form a slug flow accompanied by a cyclic flow behaviour that was a characteristic of small channels. However, the similarity of the two-phase flow regime in the chimney section could distort the drift-flux number. This was due to the fact that while the local relative velocity itself remained prototypic, the velocity of the reference inlet is scaled by a different ratio. The distortion in drift-flux number led to distortion in the void fraction.

There are more examples of scaling distortion in the experimental test facilities, e.g. Bessette & Di Marzo, 1999. Section 3.2.6 "Scaling distortions in experiments" systematically explores the root causes of these distortions. These sources of distortion include the heat loss, boundary flow, pressure drops in the loop, multi-dimensional phenomena, the coolant pump, fuel simulator, and other localized phenomena.

They are more or less due to limitations in the scaling models used in the design. These limitations are detailed in the following section.

2.2.2 Deficiencies in scaling methods

As discussed earlier, scaling distortion occurs due to the inappropriate design in, and operation of the scaled model that which is constructed and operated according to the scaling ratios derived in the scaling method. In the science and technical communities, commonly used methods are linear scaling, Carbiener & Cudnik, 1969, power/volume scaling, Navahandi et al., 1979, and Ishii's scaling, Ishii & Kataoka, 1983. In 1991, the USNRC issued a generalized scaling methodology, the Hierarchical Two-tier Scaling (H2TS) methodology, Boyack et al., 1991. Employing this methodology, Zuber and his coworkers tried to consolidate the concepts of all previous scaling methods, and developed a comprehensive procedure for scaling. Their approach has the advantages of being logical, comprehensive, and traceable, and has been well accepted and applied in the industry. In the last decade, some new methods have become available, e.g. the FSA scaling method, Zuber et al., 2007, and the Dynamical Systems Scaling, Reyes, 2014. These new methods aim at improving on the common disadvantages of previous techniques, e.g. the qualitative approach of ranking important processes, over the use of experiments and computer simulations, and distortion quantification. It is well recognized that a complete similitude cannot be achieved, particularly in a complex nuclear-reactor system. Therefore, it is important to prioritize the similarity of the processes of the greatest interest between the prototype and the model. The assumptions and compromises made in the scaling method may distort less-important processes. Therefore, the designer needs to choose the best scaling method according to the experimental objectives.

In the linear scaling method, Carbiener & Cudnik, 1969, the dimensions of the prototype are proportionally reduced by a scaling factor, traditionally the characteristic length-ratio. The model thus constructed literally is a miniature replica of the prototype. One advantage of this method is its better interpretation of the component's interactions. Also, the transportation time of fluid and sound is proportionally scaled. However, with the time being scaled, the velocity is assumed to be preserved, which implies that the fluid's acceleration and energy transfer will be distorted. This approach is not practical, Kiang, 1985. Another obvious disadvantage is the acceptability of the scaling factor's low limit. Some processes of interest may behave differently, or even disappear as the scaling factor becomes extremely low, particularly those ones that are sensitive to the length scale, e.g. entrance effects, and boundary-layer phenomena. Furthermore, some energy-transfer processes are difficult to implement in the scaled model if the components already are physically small in the prototype, for instance, the fuel rods and steam-generator tubes.

In the 1970s PWR ECCS experiments, USNRC, 1988a, the focus of scaling design was on the power, flow-distribution and the event timings. Linear scaling obviously was not a good choice. The so- called power-to-volume method became popular. In this method, the fluid-volume scaling factor (Vr, defined as Vmodel/Vprototype) is set equal to the power ratio between the prototype and the scaled model. To preserve the time scale, the height of the facility normally is preserved and the corresponding fluid volumes were scaled according to the power ratio. As expected, the time scale, fluid mass and energy distribution, flow velocity, and other rate-dependent phenomena also were preserved in the scaled model.

However, to preserve the volume scaling, flow resistance in the pipes is compromised because the flow area is overly reduced, thus distorting the pressure drop across the component. Some remedies were used to address this issue, e.g. enlarging the pipe's diameter or shortening the horizontal pipes in the scaled model. Due to preservation of the height and volume ratio, the ratio of structure area to fluid volume is enlarged (Vr-1/2), causing the transfer of excessive heat. Atypical disadvantage is the problem of excessive heat-loss in the scaled model. The problem of the heating of the downcomer wall, discussed in Section 2.2.1, is another well-known example of distortion in volume scaling.

Another side effect of preserving the height and volume is the reduction of the flow area, which makes the flow one-dimensional. Multi-dimensional phenomena are then compromised. Important geometry-sensitive phenomena, like the ECC bypass and mixing in a downcomer during a PWR LOCA refill-phase, entrainment and de-entrainment processes in the upper plenum, and steam-binding effects in steam-generator tubes during LOCA re-flood phase are distorted. The one-dimensional flow path also significantly distorts the development of the flow regime and transition in important phases of the transient. These PWR LOCA phenomena were well investigated, using separate effect tests and qualified code calculations. The volume-scaling distortions of these known phenomena were identified, and can be well simulated by numerical experiments.

In Ishii's scaling method, the dimensions of the model are a function of the height ratio, and the designer can choose to maintain the model's height by setting the ratio to 1.0, or reducing the model's height by setting it at less than unity. This method greatly alleviates the construction costs. The method also tries to address important local phenomena by introducing several non-dimensional groups describing the important phenomena. These parameters are set as equal between the prototype and the model, e.g. friction number, modified Stanton number, heat source number and Biot number in single-phase flow; and sub-cooling number, Froude number, phase-change number and drift-flux number in a two-phase flow. This method generalizes both the linear scaling and the volume scaling, and has been used widely in recent facility designs.

Another contribution from Ishii's scaling work is the pioneering work of scaling hierarchy. He proposed a three-level scaling – the integral response scaling, viz., the mass and energy inventory, boundary flow scaling, and the local phenomena scaling. The first two levels of scaling are performed from top down in system hierarchy, and the last level is from the bottom up.

A disadvantage often mentioned in Ishii's method is real-time scaling. Based on the derivation in single- phase flow, the time-scale depends not only on the length-scale ratio but also on the power ratio. However, in two-phase flow, the time scale depends only on the length scale ratio. A complex process involves both single-phase flow and a two-phase flow, so real-time scaling is not possible for reduced length scaling, Kiang, 1985, and Ishii, 1998. But this might not be considered a major disadvantage. In the case of the PUMA long-term LOCA test, reduced height actually was deemed as an advantage. The PUMA (SBWR) facility reduced the height from the prototype SBWR by a factor of 4, and then the velocity was reduced by a factor of 2, the square root of 4. Therefore, the time was reduced by a factor of two – the length-scale ratio divided by the velocity-scale ratio. For a long-term cooling experiment, PUMA was able experimentally to complete a 16-hour SBWR transient only in 8 hours. Besides, in use Ishii's methods, the option remains to preserve the full height in design, which leads to real-time scaling. Another disadvantage is the scaling laws derived in single-phase flow do not completely satisfy the non-dimensional parameter requirements, Kiang, 1985. Thus, some distortions are embedded in the scaling process.

The aforementioned H2TS methodology is a generalized procedure using a characteristic time-ratio as the non-dimensional group (the Pi group). Different phenomena (involving the transfer of heat, mass, and momentum) are characterized by their own characteristic time ratios. The methodology formalizes the hierarchy of scaling with a top-down step dealing with integral responses and a bottom-up step handling localized processes. It emphasizes the importance of bottom-up scaling that complements the integral similitude. The relative importance of local phenomena depends on the experimental objectives, and can be determined by the characteristic time ratios of the processes. The methodology has been applied in several scaling-analyses of new reactor designs. One issue became evident with this methodology is when the scaling ratios derived from the non-dimensional group (the Pi group) in the top-down step and the bottom-up step conflict each other. The designer needs to choose the scaling ratio to minimize distortion by comparing the relative importance of the competing factors in the same period of the scenario.

Another issue of this methodology is the definition of distortion quantification. A relative difference-ratio of the characteristic time ratios (the non-dimensional groups) was defined as the quantified distortion.

In reality, this ratio could vary greatly, from -200% to +200%, due to the large range of the non-dimensional group. However, the methodology also states that distortions only are important when the underlying phenomena have large impact on the figure-of-merit. The scaling methodologies do provide the ranking of the phenomena for each type of integral-balance equations. With large difference in values of non-dimensional groups that represent significant phenomena, the facility will be considered with large distortion, and hence, may not be useful for the application. The top- down approach assures that figure of merit is considered in the distortion. Figure of merit could be the vessel's inventory or the peak clad-temperature.

In the Fractional Scaling Analysis (FSA) method, the effect (or fractional change) of the state variable (or figure of merit) by the agents of change (AOC) can be derived from the existing experimental data and analyses. It also offers a quantification of the scaling distortion by the difference of fractional changes between the model and the prototype. More details of FSA method are given in Section 3.1.7.

FSA has been applied to design new facilities and to assess the scaling of existing facilities, Da Silva et al., 2009, Blandford, 2010, Aydemir, 2009, Botelho et al., 2010, and Carelli et al., 2009. These facilities simulate a variety of reactors, namely, CANDU-6, IRIS, and the Advanced High Temperature Reactor. In their original paper, Wulff et al., 2005, an important feature of FSA, the synthesis of information, was demonstrated. This synthesis could reduce the effort to address different transients of same class.

FSA evolved into DSS as discussed in section 3.1.2.8.

Here, it should be noted that the key objective of the scaling methods is to design test facilities as well as the related test conditions. Scaling methods are essential tools in the area of nuclear thermal-hydraulics facilities. This also is described in Section 3.1, discussed in Section 3.2, and the qualification of the methods by the counterpart tests is discussed in Section 3.3. Then, the following four drawbacks or limitations may still remain in the application to nuclear reactor safety (NRS):

a. Choice of starting equations.
b. Approximations in selecting non-dimensional numbers for scaling some local phenomena.
c. Details of the geometry and initial conditions of an NPP.
d. Local validity.

(a) Choice of starting equations

Balance equations are used in the scaling process in both the top-down and bottom-up steps. They may be local equations, but generally are simplified (e.g. using thermal equilibrium) and integrated over a control volume. Contributions from convection, diffusion, and volume sources have to be estimated and models often are necessary. The quality of the scaling strongly depends on the validity of the simplifying assumptions, and of models used for agents of change. Relevant points are discussed in the next section and in Section 4.1.

(b) Approximations in selecting non-dimensional numbers

In the bottom-up approach, local phenomena are scaled by using identified non-dimensional numbers that supposedly control the basic process. However in two-phase conditions, it is extremely difficult to select these numbers before knowing the flow regime that is part of the system's response during an accident transient and the local parameters of the flow in all phases of the transient. The wall and interfacial transfer terms are approximated based on average fluid conditions, for example. Relevant points are discussed in the next section and in Section 4.1.

(c) Details of the geometry and initial conditions of an NPP

Some details of nuclear power plants or test facilities are not known by the analysts. Those details include sharpness of the edges at any branch, heat losses, and spatial distribution, size of dead-ends, small leakages e.g. from valves, the initial distribution of fluid temperature in stagnant- or nearly stagnant-regions like the PWR upper head. Some of these may have decisive influences on the evolution of selected accident scenarios of interest. However, if these details are known and the PIRT showed that pressure drops there are important, they can be addressed by further breaking the control volume, and these additional losses are at the boundary of the control volumes. In most cases, they are lumped into total frictional losses. Unfortunately, none of those parameters can be firmly addressed by scaling factors, nor have they been used so far within the application of scaling methods. The relevant points are discussed in Section 3.2. The effect of a sharp edge in the pressure drop at geometric discontinuities with possible local cavitation is discussed in Section 3.2.6. Concerning the effect of number of nucleation sites on the TPCF, no scaling factor has been proposed and it is difficult to use a prototype water condition in scaled experiments.

(d) Local validity

Fulfilling scaling-driven, or a scaling method originated dimensionless quantity (e.g. the value of Re) is possible and straightforward at one time, i.e. under steady state condition at one location. In NPP accident analysis, phenomena that are occurring at a huge number of locations are of interest at the same time and in each location. Furthermore, transient conditions occur, and time derivatives may be important, see item **(b)** above. This issue is discussed further in Sections 3.1 and 3.2. The following are two examples:

1) In connecting a large horizontal pipe (e.g. HL and CL) and a smaller sized branch pipe (surge line, ECCS line, or break pipe), the relative position of the two axes is decisive for the evolution of some thermal-hydraulic phenomena, such as vapor pull-through and liquid carry-over. Such phenomena can be predicted by theoretical- or empirical-models, e.g. Smoglie et al., 1987, Maciaszek, 1987, Yonomoto & Tasaka, 1988, and Maciaszek & Micaelli, 1990, and could be scaled using them. However such scaling may be in conflict with other scaling criteria that are used to define the diameters of the main pipe and its branches. For example, the evolution of quality in a break pipe (diameter d) at the top of a horizontal leg (diameter D) depends on the relative height of the liquid pull through H_{lim}/D. In a power-to-volume- scaled IET with a volume-scaling factor λ, d must be scaled with d multiplied by $\lambda^{2/3}$, and D often is multiplied by $\lambda^{2/5}$ to preserve the Froude-number similarity. According to the H_{lim} model by Maciaszek, 1987, H_{lim} is multiplied by $\lambda^0 = 1$ and H_{lim}/D is not preserved.
2) Heat transfers of a given piece of space with distorted geometries, e.g. different scaling factors for horizontal- and vertical-dimensions, may then induce distortions of some natural circulation (NC) effects that cannot be quantified by scaling methods. D'Auria et al., 1991, presented differences measured at a system level in case of NC in a PWR ITF that: a) are originated by various distortions in the design, and b) cannot be quantified by applying scaling methods.

2.2.3 Deficiencies in system codes and CFD

In NPP safety analysis computer codes are commonly used because of cost and time effectiveness. Before a computer code is used for safety analysis, the user needs to address the question – is the code applicable for the analysis? Without sufficient justification, the results from the analysis may be erroneous and misleading.

The computer codes commonly used in the safety analysis are categorized into two groups - The system thermal hydraulic (SYS TH) codes, and computational fluid dynamics (CFD) codes. Both categories of codes are not first-principle codes, but CFD codes are closer than the SYS TH codes in general. The CFD codes also require same empirical relationships for the interfacial transfer of mass, momentum, and energy. They are only closer to first principle for single-phase flows.

The empirical models used in the system codes to represent the terms in the field equations generally are developed from experiments wherein the geometry and the boundary- and initial-conditions were not typical NPP-operating conditions. Many of them were not even intended for nuclear reactor applications, but currently are used in the nuclear systems. For example, the Dittus-Boelter correlation, (Dittus & Boelter, 1942), which was developed from experiments for automobile radiators of the tubular type, now is widely accepted in the computer codes for modelling turbulent heat transfer, regardless of the geometry. It certainly raises the issues of applicability. If the models were formulated in physically meaningful non-dimensional parameters (e.g. the Nusselt number, or the Froude number), the empirical models usually are more extendable to full-scale applications provided that the operating range covers the NPP's conditions, and there is no change in flow configuration, such as the flow regime or the developing flow. Otherwise, the fluid behaviours and energy-exchange processes in the NPP could deviate greatly due to different flows, flow regimes, and interfacial-transfer processes. In addition to empirical models, sometimes tuning constants were used in the validation process to ensure better agreement between the test data and the calculation. These tuning constants, e.g. flow resistance coefficients, and heat transfer fouling factors, could cover up the distortions from inter-acting empirical formulas. These tuning constants generally are not scalable and require specific evaluation.

Most CFD applications in the nuclear industry are single-phase simulations. The application in two-phase flow still is limited due to lack of maturity of physics laws and computing power. Similar to system codes, scalability issues exist in applying CFD. There are several physics models (k-ε, k-ω, RST, SST, RNG, wall laws) and numerical schemes in the CFD codes. Extending the choice of physics models and numerical settings from a scaled experiment to nuclear-reactor application requires justification. Nodalization (grid) is another subject that requires investigating since the physical size in the prototype usually is much larger than the scaled test facility. Appropriate grid size and arrangement need to be evaluated in accordance with the Best Practice Guideline, NEA/CSNI, 2007. A reasonable question arises if the nodalization of the model differs from that of the prototype and yet both follow the best practice guidelines. Another challenge in CFD simulations is the evaluation of scaling uncertainties. Usually, most methodologies involve many calculations that could be impractical due to high CPU cost, NEA/CSNI, 2007.

The numerical method used in the code also is a source of scaling distortion. In validating the scaled experiment, different options and constants of the method are optimized to produce the best agreement. These numeric settings are not necessary transferrable to NPP simulations because of larger physical size of the NPP and design differences. There will be no opportunity to benchmark the settings since usually no full-scale data are available.

In numerical simulations, the flow- and energy field-of interest are discretized for numerical solutions. Within a node (or a cell), the fluid properties are assumed to be the area-(or volume-) averaged values. The "averaged"-fluid-property approach, particularly in the coarse scheme of NPP simulations, could fail in not capturing the expected behaviours of the fluids due to limited resolution. On the other hand, a finer nodalization could introduce a violation of the Courant limit issue. Therefore, these two factors need balancing in determining the nodal size. The arrangement of cells (nodalization) is an important task in system code simulation to capture important processes. Incorrect representations or an overly simplified arrangement of nodes could disable processes of interest, or distort the fluids' behaviours. Similar to the numeric settings, it remains questionable when an optimized nodalization in a scaled experiment can be extended to an NPP simulation. And yet the same numeric- and nodalization-scheme are preferred to avoid unexplainable results.

2.2.4 Quantification of scaling distortions

As discussed earlier, it is very unlikely to attain perfect similitude between the prototype and the model for all phenomena and processes in a transient. The common practice is to optimize the similitude of phenomena of the greatest interest, which usually accompanies distortion in the less important processes. The impact on the transient by these distortions needs to be evaluated and justified before the constructing the model. In Section 2.3.5, the experimental design will be elaborated upon.

In the scaling analysis, the scaling ratios usually are derived from non-dimensional parameters. These parameters are set to be equal between the prototype and the model. But the reality is that, for practical reasons, the final constructed dimensions will deviate from the theoretical values, this means that the non-dimensional parameters of the model will differ from those of the prototype. It is natural to judge the distortion by evaluating the difference of the non-dimensional parameters of the prototype and the model. Acceptability is based on a tolerable criterion for the difference. A well-accepted criterion for scaling distortion remains controversial in the international nuclear community. The level of distortion that is acceptable is based on the application of the tests. The requirements are less rigorous for validation, but lot more so when the findings are used for quantitatively estimating uncertainty.

In H2TS, Zuber and coworkers (Zuber et al., 1991) used the characteristic time ratio of a process as the non-dimensional parameter for scaling. It is defined as the ratio of the residence time of the process quantity in the control volume, and the characteristic time of a particular process. The characteristic time refers to the time required for a complete transfer of the quantity (mass, momentum, or energy) in the control volume. Therefore, the characteristic time-ratio denotes the quantity changed by the specific process to the total quantities available in the control volume. A larger characteristic time-ratio means more quantities are changed by the process, and therefore, the process is more active in the transient. The H2TS methodology proposed using as the distortion the percentage difference of the characteristic time ratio between the prototype and the model. This distortion value represents the percent difference that a specific transfer process changes the reference quantity during its residence time in the prototype and model.

Due to the huge range of the referenced parameters, the resulting non-dimensional parameter (or characteristic time ratio) could have a large range. The quantified distortion could fall into a large range, say -200% to + 200%. Hence, it is difficult to compare the relative importance of the processes, and to determine the acceptability of the scaling design. On the other hand, it is desirable to know the direct relationship between the parameter of interest (i.e. the figure of merit) in the transient and the distortion. The available quantification methods for distortions do not provide this linkage.

As discussed later, the propagation of the effects caused by distortions raises another need to call for a method that can evaluate the accumulated distortion of a particular process as a function of time, not just its distortion evaluated at a particular time.

2.2.5 Propagation of scaling distortions

As discussed in Section 2.2.1, scaling distortion is a significant source of uncertainty. This distortion, depending on the scaling methods and the targeted phenomena, could occur in certain components at a certain time in the process. Evaluation of the overall scaling distortion is more representative than scaling distortion evaluated on a particular component in a particular phase of the transient. In evaluating the overall impact to the parameter of interest, distortions from all the system's components in each phase of the transient should be accounted for. In the CSAU methodology, the overall uncertainty includes all possible sources of uncertainty in all phases of the scenario because the effects of the uncertainty on the parameter of interest could propagate from one phase to the next, and so accumulate. Similarly, the effects of scaling distortion could be passed on to the future phases of the scenario and change the overall scaling distortion.

The PWR LOCA scenario is a good example to illustrate this concept. In the scaled test-facilities for a PWR LOCA, the power to volume-scaling method has been mostly used. In Semiscale, to preserve the core's height, the flow paths of the entire test facility were constructed close to one dimensional, and the downcomer is surrounded by an excessively heated wall. As discussed in Section 2.2.1, during the refill process the excessive wall heating due to scaling distortion in the downcomer delayed the ECC reaching core inlet. This delay moved the phase boundary between the refill phase and the re-flood phase, obviously affecting the PCT. During the reflood phase, the nature of the one- dimensional flow in the upper plenum could not represent the multi-dimensional flow pattern in the prototype. Thus, some important phenomena were distorted, like entrainment, de-entrainment, and steam binding. These distortions certainly affected the PCT results as well. Therefore, the effects of all scaling distortions in each phase accumulate and are reflected in the final value.

Therefore, in estimating the overall distortion, a new method is needed for quantifying the time-dependent scaling distortion. This quantity should include all the distortions that occurred during the transient. In current approaches, the distortion definition is evaluated based on the parameters at a certain time in the event. The distortion thus obtained can only represent the distortion in that moment of a particular phase although these distortions are used as representative of particular phase of the transient. Using a time-dependent method of quantification to represent the overall distortion could mislead the experimental results because the parameters of interests are not just affected by one particular scaling distortion, but also by others.

2.3 Scaling in complex phenomena and test design

2.3.1 TPCF – two phase critical flow

Large classes of design-basis accident scenarios involve some type of outflow from the reactor system, either through a pipe break or valves that are stuck open. The flow could be choked at high pressures. This could be subcooled choking, where flashing occurs at the break, or saturated choking where the flow at the break could be a two-phase mixture. The computer codes for reactor simulations generally separate the correlations for subcooled- and saturated-choking. These models predict the area-average flow rates through the break, based on the conditions just upstream of the break. As these critical flow-models are based on fluid properties and thermal-hydraulic conditions (pressure, temperature, and void fraction), they are expected to predict the break area's average mass flux as independent of scale for application with the same fluid. However, non-fully developed flow conditions occur upstream of the break and the development of the flow depends, almost unavoidably, upon scale. Lack of fully developed flow will also impact the critical flow. In the case of power-to-volume scaling, the break size is scaled to volume to preserve the time scale. Ishii et al., 1998, also stated that the critical velocity or mass flux is dependent on the property and thermal hydraulic conditions, and this scaling should be applied to the design of the break size.

It is understood that in two-phase choking the critical flow will depend on rate of vapor generation at the choke point. The compressibility of the mixture will depend on the rate of vapor generation, as fluid particles move towards the break. The rate of change of density with pressure is the inverse of sonic velocity, and, in case of two-phase flow, there is large change in the density of the mixture with the change in pressure, even more than in the gas phase. The sonic velocity in two-phase flow is lower than that in the single-phase gas flow. This vapor generation will depend on the flow regime, interfacial area, interfacial momentum-transfer or slip, and liquid superheating or the degree of non-equilibrium. There are no tests that measure local details of two-phase flow in critical flow-tests.

There are few documents, viz., Saha, 1978, Elias & Lellouche, 1994, Levy, 1999, Sokolowski & Kozlowski, 2012, and NEA/CSNI, 1980, which describe various models and tests that are available for critical flows. Elias & Lellouche, 1994, listed 66 tests with different lengths, diameters, and pressures. However, only 42 data sets were usable. The diameter of a broken pipe varied from very small at 4 mm, to large at 76 mm. The length varied from no pipe to a 1,700 mm pipe. The Marviken Project, 1974, provided a data set that was much closer to the reactor system. The pipe diameter varied from 200- to 500-mm, and length varied from 166- to 1800-mm.

There are critical flow models based on various assumptions about thermal- and mechanical-equilibrium. The simplest model is a homogenous equilibrium model wherein two phases have same temperature (saturation) and same velocities. There are other correlations that relax one of two equilibrium conditions. There are models, such as thermal equilibrium with slip, and thermal non-equilibrium without slip. Elias and Lellouche (1994) study indicated that none of the models did well with the data. Scatter plots indicated that most of the data was outside ±10% of the mean. Saha (Saha, 1978) earlier reviewed various models and concluded the following.

1. The Homogeneous-Equilibrium Model (HEM) under predicts the critical flow-rates for short pipes and near-liquid saturation, or subcooled upstream conditions.
2. The equilibrium-slip models of Fauske, Moody and others, although successful for long tubes, under predicted the critical flow rates for short pipes. This is particularly true if the upstream condition is subcooled, or near saturation.
3. The effects of thermal non-equilibrium must be taken into account for short pipes. However, it is not clear whether the pipe length, L, or the pipe length-to-diameter ratio, L/D, or both, are important in determining the effects of thermal non-equilibrium.
4. At present, there is no general model or correlation for critical flow that is valid for a broad range of pipe lengths, pipe diameters, and upstream conditions, including subcooled liquids. The more sophisticated the model used for a more precise design, the more important it is to consider some data under similar conditions.

An assessment of RELAP5 and TRACE critical flow models was also undertaken (Sokolowski & Kozlowski, 2012). Their conclusion was that a code's predictive accuracy was independent of the length of the pipe before the break. Another conclusion implied that same correlation worked better with one code than the other. These results indicated that the effect of length alone is not conclusive. This also means that two other phase-flow models that estimate the vapor generation may have a more important influence on the critical flow. Also, the diameter effect has not been established. However, unless cross section flow variation is significant, the diameter effect will diminish with the size of the pipe. The fact that length affects the level of thermal non equilibrium at the break and thus affects the flow has been accepted. Therefore the scaling plays an important role in determining the critical flow.

There has not been any systematic scaling study for critical flow phenomenon. The sub phenomena that affect the critical mass flux were identified, but the data have not been compiled with dimensionless groups representing sub-cooling, flow rate, slip, wall friction, nucleation sites, the geometry of the break, and two-dimensional effects.

There have been few studies that account for nucleation and bubble growth models to estimate vapor generation and its effect on critical flows, Rohatgi & Reshotko, 1975, and Richter, 1983. While these studies provide a framework for modelling, they require a free parameter, such as nucleation site density, and it is not yet measured quantitatively. It may vary by several orders-of-magnitude, typically ranging from 109 m-3 to 1011 m-3. These analyses have used friction loss at the wall. In the case of a gas flow with wall friction, the critical flow is estimated from the model and it is called Fanno equations. The reason for decrease in critical flow for the gas flow is that stagnation pressure decreases, and with that, critical flow decreases.

The system codes calculate conditions at the break plane based on (code) models for the wall and interfacial transfer terms that include wall friction. However, flow conditions at the basis of the development of those models are far from critical two phase conditions (e.g. fluid velocities and slip ratio). Furthermore, these system codes include constitutive relationships based on steady-state data while the flow conditions near the choke point are rapidly changing. Therefore, imposing a critical flow model to predict break flow or criteria to establish possible changes in the location of the choked section may not be sufficient for a suitable prediction of expected phenomena.

At this point, Marviken tests will provide close-to-full-scale data that can be used for code validations and uncertainty analyses.

2.3.2 CCFL – counter-current flow limitation

The countercurrent flow of gas and liquid appears in many places in the reactor system, such as the steam generators, the core upper tie plate, hot and cold legs, and the downcomer during phases of various transients, (NEA/CSNI, 1993). In most cases the liquid flows down and gas flows up, and they interact at the interface. The shear at the interface that increases with the gas flow rate opposes the liquid down flow. This limits the liquid down flow. Countercurrent flow limitation is the limit of this liquid flow due to the opposing gas flow.

The upper plenum tie-plate is a plate with holes. Similar plates exist in BWRs and PWRs. In both cases, liquid may accumulate on their top, and flow down on the top of the core. However, the upward flow of steam may impede or prevent this down flow of liquid. The upper support plate's flow is multidimensional because the steam's up-flow is not uniform; the correlations generally are for average of flows over the whole plate. There are data available at full geometric scale from UPTF, Emmerling et al., 1988, Glaeser, 1989, Glaeser, 1992, and Glaeser & Karwat, 1993.

One of the most important phenomena for LBLOCA safety is the emergency core-coolant bypass, where some of the coolant injected in the cold legs during LOCA, is bypassed to the broken cold leg instead of going down in the downcomer and filling the lower plenum, and flooding the core. This bypass is caused, in part, by the steam flow from the core to the broken cold leg through the downcomer. This steam up flow in the downcomer impedes, and in some cases, reverses the downward flow of coolant from the intact cold legs. The downcomer has countercurrent flow where steam flow transfers momentum at the interface, and also creates waves and entrainment at the interface. The combination of these phenomena leads to a limitation in the down flow of coolant for a given up-flow of steam, coolant sub-cooling, and downcomer geometry. It also was observed that in the downcomer type geometry, the liquid coming from cold legs spreads azimuthally in the downcomer. This spreading phenomenon is much different from the counter-current flow in pipes.

There have been many experimental studies at different scales to address CCFL (Levy, 1999). They are in pipes and in annuli. Tests in pipes of different sizes, and of methods of injecting liquids were used to develop CCFL correlations by Wallis (1969) and Kutateladze (1951). The Wallis correlation is based on the pipe diameter as length scale. Non-dimensional volumetric fluxes for gases and liquids are well correlated. However, this correlation diverges from data for pipes of larger diameter pipes.

$$j_g^* = j_g\sqrt{\frac{\rho_g}{gD(\rho_l-\rho_g)}} \tag{2-1}$$

$$j_g^{*1/2} + m j_l^{*1/2} = C \tag{2-2}$$

For pipes with diameters greater than 5 cm, the CCFL can be represented by Kutateladze correlation. It has length scale based on surface tension. This correlation applies to large pipes where waves at the interface are smaller than the pipes' diameters.

Traditionally, the Wallis model is used in smaller pipe components. However, experiences from the analysis of UPTF data, Wolfert, 2008, and Mayinger et al., 1993, show that it can be extended to larger pipe as well, like the hot leg of PWRs. The Wallis model predicted the reflux of the condensation return flow back to the hot leg well in both the high- and low-pressure range.

There have been correlations similar to Wallis correlation for annular flow, but they were not widely used. A systematic scaling study was done for annular geometry, Yun et al., 2004.

Rohatgi et al., (CSAU, Table 3.1, 1990), reviewed different tests with annular geometry. These tests were UPTF (1/1), CREARE (1/5), BCL (2/15), BCL (1/15), CREARE (1/15) and CREARE (1/30). These facilities were designed with linear scaling. A comparison of the TRAC-PF1 prediction with the data, Rohatgi et al., 1990 (Fig. 3.12), indicated that TRAC-PF1 over-predicted the lower plenum filling rate (and under predicted the ECC bypass) for smaller facilities. For the UPTF tests, TRAC-PFI under-predicted lower plenum filling rates. Analyses of the UPTF data indicated that most of the injected ECC was unaffected by the steam flow.

Yun et al., 2004, showed that the linear scaling method is not appropriate, and provided a modified linear-scale method. The facilities for studying CCFL phenomena are of annular shape. The flow is multidimensional, as it spreads azimuthally and thickness of the film is important. The annular gap affects the flow regime expected in the downcomer; therefore, its size should be such that the expected flow regime is preserved. The film's thickness as it spreads under the cold leg in the downcomer is same fraction of the downcomer gap as that expected in the plant. It implies that the interfacial shear is same. The downcomer's aspect ratio, that is the gap and circumference ratio, must be preserved as it affects the film's spreading. Yun et al., 2004, also showed that data plotted in the Wallis form of superficial-gas velocities at different sizes match.

In PWR, small and intermediate cold-leg break LOCAs, loop seal clearance is important in determining the period over which the core was uncovered, along with the flow-pattern and depressurization scenarios. The returning condensate from steam generator in the reflux condensation mode, and the ECCS injection coolant tend to accumulate in the loop-seal section and stop the flow path of steam generated in the core due to the CCFL phenomenon. The pressure drop between the core and the leak through the hot legs increases so that upper plenum-downcomer-cold leg bypass paths are created to relieve the pressure. Detailed data is needed to explore the clearance phenomena in the loop seal, and for system code benchmarking. Some scaled-down- and full-scale-experiments using an air-water mixture under atmospheric pressure were carried out in trying to correlate the data. In the UPTF program, Liebert & Emmerling, 1998, ran a full-scale steam-water mixture experiment to study integral effects using four loops, and the separate effects using one loop of the loop seals' clearance process. The main conclusions from the integral- effect test conclude that the loop seal clearance sequence in the 4 loops depends on the break size, and that the pressure drop of UPTF in the bypass flow path is different from that in a real PWR due to geometric differences. Tests of the separate effects show that the measurement of the level of residual water after clearance of the loop's seal agrees with the Ishii- and the modified Kutateladze-formulations, which can be extended to the condition of the real reactor. The flow pattern of the loop-seal clearance observed in the UPTF experiment agree partially with the Taitel-Dukler transition criterion between stratified flow and slug flow; the observation is not extendable to the reactor's condition. The pressure drop across the loop seal is much higher in the steam-water mixture than in the pure steam flow and differs greatly from those of the scaled down air-mixture experiments.

It is recommended that UPTF data or any other full scale data be used for validating the code and for uncertainty analyses as they are available.

2.3.3 Entrainment and De-entrainment

A nuclear-reactor system has entrainment- and de-entrainment-phenomena taking place in different components, such as the core, the steam generators, the steam separator in a BWR, and the suppression pool. There are three scenarios - one in which the gas bubbles out from the interface, the second, wherein the gas flows parallel to the interface, and the third, where water is separated by centrifugal forces, such as in the BWR Separator/dryer and the steam generator's secondary side.

The phenomena of entrainment and de-entrainment are important as they affect the distribution of liquid in the gas phase, and affect the interfacial transfer of energy, mass, and momentum through the shape and size of the interface. These phenomena occur at the interface between gas and liquid phases. The mechanism for entrainment and de-entrainment differ. Entrainment occurs due to relative velocity between gas and liquid, leading to instabilities at the interface. The peaks in the waves are broken by the gas phase, and result in liquid entrainment. This captured liquid may break up further till a stable drop size is achieved.

There are many mechanisms of de-entrainment. In the case of the horizontal flow of gas and droplet over an interface, any decrease in air velocity will lower the shear between the gas and droplets, leading the droplets to be deposited on the interface. In vertical flows, there is the continuous impingement of droplets on the interface, along with entrainment, created by the wavy interface. As gas slows down, the waviness will decrease, as will entrainment decrease, with the net effect of de-entrainment exceeding entrainment. In addition, if there are many steps or barriers in the direction of flow, the droplet will impinge on them and separate from the gas-droplet mixture. While gas can change direction, the droplets, due to their larger momentum, cannot easily change their direction and impinge on the barriers. Grid spacers in the core act as a separator for droplets. These droplets, impingent on the solid surfaces of heated rods during accidents, provide precursor cooling prior to a full-scale reflooding. The phenomena of entrainment are at the localized process-level, and generally are not modeled by first principles in the system codes. Instead, the codes use correlations for net entrainment.

The system codes, such as TRACE and RELAP5, have correlations for the entrainment fraction, E_∞ based on Ishii and Mishima's correlation, (Ishii & Mishima, 1989):

$$E_\infty = tanh\left[7.25 * 10^{-7} We_g^{1.25} Re_f^{0.25}\right] \tag{2-3}$$

where,

$$We_g = \frac{\rho_g j_g^2 D_h}{\sigma}\left(\frac{\Delta\rho}{\rho_g}\right)^{1/3} \tag{2-4}$$

$$Re_f = \frac{(1-\alpha)\rho_l V_l D_h}{\mu_l} \tag{2-5}$$

As can be seen, entrainment is based on the local Weber number and Reynolds number, and depends on the diameter of the tube. The interface instabilities generally depend on the thickness of the film. The film's thickness will be larger for larger diameter pipes for same void fraction, and is more likely to have instability at the interface.

The other mechanism of entrainment is when gas bubbles emerge from the interface, and as they leave it, a liquid filament is created that breaks and leads to the formation of drops. The drops have some momentum from the gas phase. However, some of the drops can fall back to the interface. A good review of this type of entrainment was given by Bagul et al., 2013, and Ishii & Kataoka, 1999. As in this situation, the liquid phase is stagnant except for activity at the interface, and the entrainment ratio, E_{fg} (h, J_g), is defined differently, Fig. 2-1 below. It is the ratio of the gas phase supplied and the liquid entrained above the interface. The ratio of liquid entrainment and gas flow is shown here as function of the height of the gas space, h. Droplets can fall back into large gas spaces above the interface.

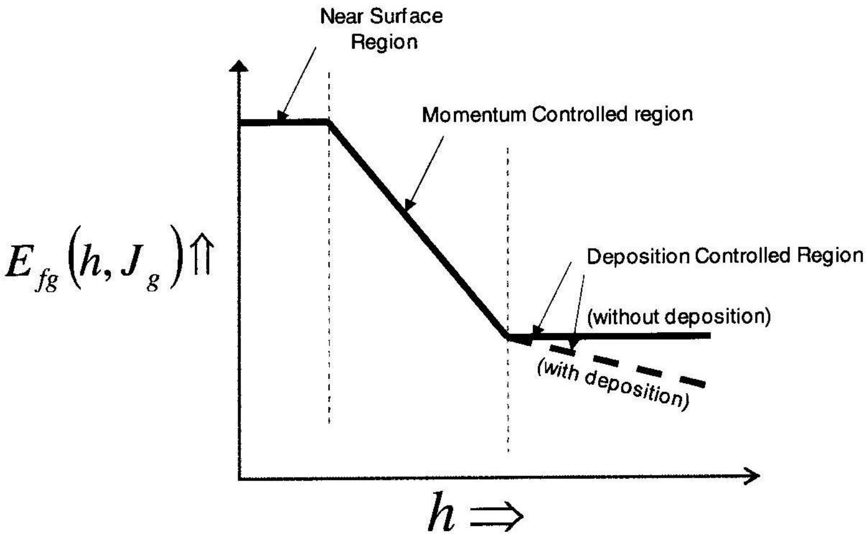


Fig. 2-1 – Regions of entrainment above the separation interface (taken from Ishii & Kataoka, 1999).

Near the interface,

$$E_{fg}(h,J_g) = 4.84 * 10^{-3}\left(\frac{\rho_g}{\Delta\rho}\right)^{-1.} \tag{2-6}$$

The deposition-controlled region,

$$E_{fg}(h,J_g) = 7.13 * 10^{-4} J_g^{*3} N_{\mu g}^{0.5}\left(\frac{\rho_g}{\Delta\rho}\right)^{-0.31} exp\left(-0.205\frac{h}{D_h}\right) \tag{2-7}$$

For the intermediate region, the entrainment depends on momentum exchange and rate that is given for three different gas-flow rates. For high gas flows the correlation near the interface applies.

For low flows, the correlation is as follows.

$$E_{fg}(h,J_g) = 2.21 N_{\mu g}^{1.5} * D_h^{*1.25}\left(\frac{\rho_g}{\Delta\rho}\right)^{-0.31} J_g^* h^{*-1} \tag{2-8}$$

For intermediate flows, $J_g^* h^{*-1} > 6.39 * 10^{-4}$, a different correlation was recommended:

$$E_{fg}(h,J_g) = 5.42 * 10^6 J_g^{*3} N_{\mu g}^{0.5}\left(\frac{\rho_g}{\Delta\rho}\right)^{-0.31} D_h^{*1.25} \tag{2-9}$$

$$J_g^* = J_g / \sqrt{\left(\sigma g \Delta\rho\rho/\rho_g^2\right)^{1/4}} \tag{2-10}$$

$$N_{\mu g} = \mu_g / \left(\rho_g \sigma \sqrt{\sigma/(g\Delta P)}\right)^{1/2} \tag{2-11}$$

An important region of de-entrainment is the steam separator and dryer in the BWR and a separator in the steam generator's secondary side. In both cases, the vapor/ liquid mixture is send through curved channels to create centrifugal forces that separate the liquid from the steam. Most codes use empirical values of carry over and carry under fractions, and from that, the amount of liquid separated from the two-phase mixture is estimated. TRACE also has a mechanistic model based on its joint development undertaken by EPRI, USNRC and General Electric. It is available in TRACE code, and it consists of the conservation of water mass, vapor mass, axial momentum, and angular momentum, entering and leaving the separating barrel. The model assumes a form of void distribution and vortex flow. The model has four parameters that are determined from the tests.

The scaling issues are the effect of pipe size on entrainment in an annular flow and due to empirical basis for obtaining carry over and carry under fractions for separators. For developing correlations or for model validation, full-sized separate effects tests are needed.

2.3.4 Reflood

The safety of nuclear fuel in a LWR is ensured by keeping the fuel covered with the coolant. In case of loss-of-coolant accidents, the reactor loses inventory that may lead to the degradation of cooling due to uncovering the fuel. The uncovered fuel will heat up, due to the decay heat. Reactors have safety-injection systems, such as emergency core coolant, high-pressure injection and low- pressure injection to fill the vessel after the system has sufficiently depressurized to allow such injections. The majority of the PWRs have injection in the cold legs, but few designs have injection in down comer and in the hot legs. The coolant flows down in the downcomer to the lower plenum, and then to the core. In case of additional injection in the hot leg, the coolant condenses vapor in upper plenum and eventually some of this coolant flows down in the lower plenum. This hot-leg injection creates an internal recirculation flow of cold coolant flowing down, and a two-phase mixture flowing through other part of the core (the central part); it prevents steam binding by condensing the vapor in upper plenum, so preventing it from going to the steam generator. Iguchi (Iguchi, 1998) discussed the effect of the distribution of radial power and of combined coolant injection.

These transients generally have three periods - blowdown, refill and reflood. The blowdown period lasts about 30 seconds, the refill period about 10 seconds, and the reflood period lasts about 250 seconds. The blowdown period ends when ECC injection begins, and the refill period ends when lower plenum is full. During the reflooding phase, the coolant starts to fill the core and quench (cool) the fuel.

The physics of quenching the core is complex. If the temperature of the clad is above the minimum stable film-boiling temperature (Tmin), the cooling will be through film boiling. However, due to axial conduction and precursor cooling, the temperature of the clad can be reduced sufficiently (below Tmin) to allow for transition and subcooled boiling. The region above subcooled boiling still is film boiling with inverted annular flow-regime and a dispersed-droplet regime created from the breakup of the core of the inverted annular regime. The quench front or transition to a subcooled boiling region moves up the core as the steam and droplet cool the core ahead of the front, along with axial conduction. USNRC studies have identified the phenomena, and their impact on heat transfer from the clad (Hochreiter et al., 2010, and Odar, 2001). There are two different regimes during the reflood period, based on inlet sub-cooling, and the coolant’s velocity, as shown in Fig. 2-2.

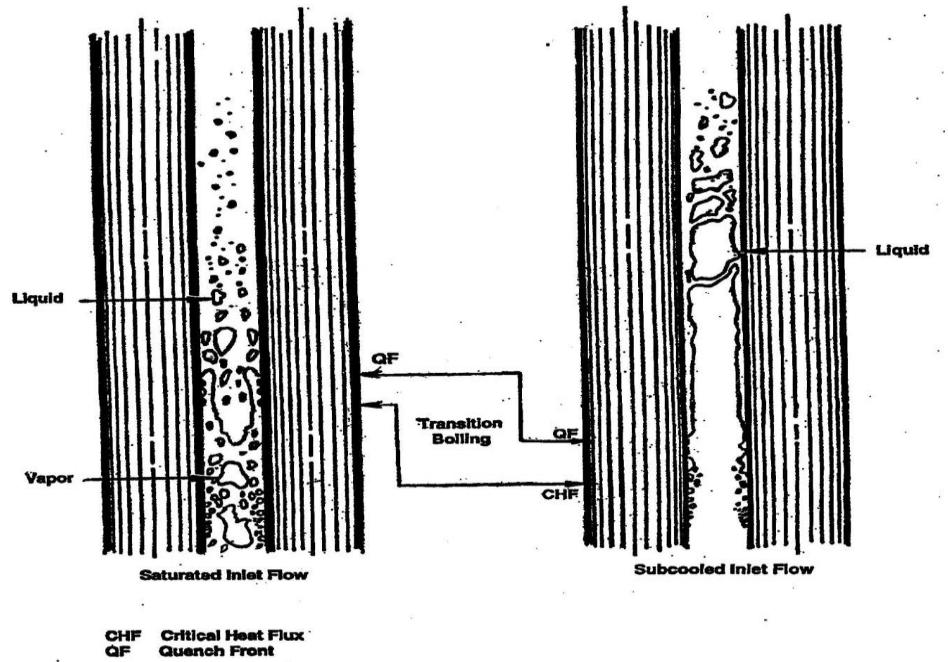


Fig. 2-2 – Qualitative reflood flow regimes at different sub-cooling (high on the right side) and flow velocity (low on the left side).

In the cases of saturated or low sub-cooling, or low-velocity core-inlet flow, the rate of vapor generation rate is high, and flow regime changes from saturated boiling to dispersed boiling heat transfer. For high sub-cooling or higher velocity flows at the core inlet, the quenching occurs sooner and generally an inverse annular-flow regime above the quench front is established. As the vapor generation increases, the central liquid region breaks up, leading to a dispersed-film boiling regime. The droplets, in the dispersed phase, cool the vapor by evaporation, and also remove heat from the clad if they impinge upon it.

The Hochreiter' s study in support of new Rod Bundle Heat transfer Facility identified the important processes needed to be investigated in the tests for accurately modelling the reflood period. It identified important processes such as those related to the local void fraction that governs interfacial heat transfer, interfacial area, the fuel/clad material, the transition-boiling heat transfer, inlet fluid temperature and velocity, liquid entrainment, droplet size, droplet impingement-heat transfer, and the dispersed-phase heat transfer. Minimum stable film boiling temperature is also important to indicate the onset of quenching. Grid effect were not rated high, but the test results showed that grid spacers improve the heat transfer due to breaking up the boundary layer and break up of drops.

There have been many tests to study this phenomenon. Most of them involved electrically heated fuel-rod simulators without gas gap. Ihle & Rust, 1987, studied the effect of a gap. It was concluded that a fuel-rod simulator without a gap exhibited delayed quenching and higher temperatures. This finding is expected as the balance of clad energy with a gap has more resistance to heat transfer from the fuel rod, so leading to an early quench.

The model for the reflood phase in the computer codes consists of correlations for Tmin, film-boiling heat transfer, dispersed- faze heat transfer, nucleate boiling, transition boiling, and conduction in the fuel rods, including the gap and the clad. All these phenomena are based on local conditions and are not subject to any scaling issue. TRACE code, USNRC, 2007, has the Stewart and Groeneveld correlation, (Stewart & Groeneveld, 1981) for Tmin, as described below. The correlation was validated with independent data. There are no terms with geometric dimensions or material properties.

$$T_{min,sat} = 557.85 + 44.1 * P - 3.72 * P^2 \tag{2-12}$$

$$T_{min} = T_{min,sat} + \frac{(h_{l,sat} - h_l)}{h_{fg}} \frac{10^4}{(2.82 + 1.22P)} \tag{2-13}$$

The convective heat transfer in the inverted annular region is based on estimating the size of the vapor film on the wall, and its enhancement due to disturbances at the vapor-liquid interface. The correlation was enhanced further to match rod bundle data. There also is a correction for natural convection in the vapor film. Recognizing that the film's thickness determines the natural convection fluid dynamics in the vapor region, and this thickness is directly proportional to the tube's hydraulic diameter. TRACE also applies a correction of a 30% increase for heat transfer coefficient based on tube data for application to rod bundles. The transition from inverted annular flow to dispersed phase is assumed to be at void fraction of around 0.6. The heat transfer in the dispersed phase is modeled with a correlation proposed by Forslund & Rohsenow, 1968. Sudo provided a set of saturated- and subcooled-film boiling correlations, Sudo, 1980. The description of TRACE models (or any other system code) shows the need for well-instrumented tests with heated rod bundles.

There have been many rod-bundle tests to study reflood phenomena, Hochreiter et al., 2010. These are listed here:

- FLECHT Cosine Tests (NRC/Westinghouse);
- FLECHT Skewed Axial Power Shape Tests (NRC/Westinghouse);
- FLECHT-SEASET 21 Rod Bundle Tests (NRC/Westinghouse);
- FLECHT-SEASET 161 Unblocked Bundle Tests (NRC/Westinghouse);
- FEBA Reflood Tests (Germany);
- THTF Rod Bundle Tests (NRC/Oak Ridge National Laboratory);
- FRIGG Rod Loop Tests (Sweden);
- GE 9-Rod Bundle Tests (NRC/General Electric);
- NRU Rod Bundle Tests (Canada);
- ACHILLES Reflood Tests (United Kingdom);
- Lehigh 9-Rod Bundle Tests (NRC/Lehigh University);
- PERICLES Reflood Tests (France).

While these tests simulate flow in the rod bundles, they have limitations (Hochreiter et al., 2010), that make it difficult to either develop correlations, or validate the models as identified in this report, and listed earlier. The tests did not measure void fractions, droplet size and velocity, and the temperatures of vapors. They also have different cladding material than that in the reactors. USNRC has supported a test programme in Penn State University's Rod Bundle Heat transfer Test facility. The design of this facility was based on the H2TS scaling approach (Zuber, 1991, Wulff, 1996). The top-down approach identified different phenomena through the transfer terms on the right side of the balance equations. The scaling estimate identified three possible scale distortions, viz., the presence of housing containing rod bundles that act as an additional heat sink, electrically heated rods and gap, and finally, clad material that has an effect on Tmin.

2.3.5 ITF and SETF design and operation

A recent NEA report, NEA/CSNI, 2001, details the qualification criteria of the facility for the validation matrix for assessing thermal-hydraulic codes for VVER LOCA and its transients. The facility's qualification criteria are useful for designing and operating the integral- and separate-effect test facilities. The qualification criteria of the quality of a facility were based on five items related to the facility's design, construction, operation, use in an international framework, and personnel qualification.

The scaling laws should be suitable for deriving the design parameters. In the case of an integral test facility, two main approaches are available, i.e. preservation of the time scale (full-height, power-to-volume scaling) versus reduction of time scale (reduced-height, Ishii approach). The first approach is the one most widely used, and is recommended in the document NEA/CSNI, 2001. There are many scaling laws and design criteria in the case of a separate test facility. Thus, the geometric dimensions of the facility (as close as possible to the prototype), the uncertainties in boundary conditions, and operating conditions were considered in the processes of qualification. A well-designed facility can be badly constructed, and some characteristics of the facility can be changed during its long- term operation. A suitably scaled and constructed facility could be operated poorly. For qualifying the operation, written procedures should be available for calibrating instrumentation, for the procedures in operating the facility, and for its maintenance. To assure the quality and completeness of the experimental data, the repeatability and the uncertainty of any test essentially should be assured and well documented. Characterization tests, such as heat losses and natural-circulation performance should be well measured and analysed. The test data should be compared with that of similar facilities, and independent groups of researchers should use the test data. Finally, the qualifications of personnel should be evaluated.

In the design and scaling of the integral- and separate-effect test, PIRT (Phenomena Identification and Raking Table) plays an important role in identifying important phenomena, Zuber, 1991, Wilson & Boyack, 1998, and Song, 2006. It is important to classify the major phases of transient phenomena, and to identify important thermal-hydraulic phenomena in components and sub-systems. The problems of scale-up capability result from incorrect scaling, incomplete characterization of the test facility, insufficient knowledge of phenomena, and scaling distortions. The design should be optimized to preserve the important local phenomena in a scaled-down facility. The scaling of an integral test facility usually has two levels. The top-down- or system-level approach focuses on integral- or global-behaviour and system interactions. The bottom-up or component-level approach focuses on individual- or local-thermal-hydraulic phenomena and processes.

The typicality of facility- and test-data should be relevant to the expected conditions of the prototype. A scoping analysis may be important, aiming at comparing the prototype's expected behaviour, and that of the facilities expected behaviours during the phase of test planning, NEA/CSNI, 2001. The same code that eventually will be validated based on the test data can be adopted for this comparison. A feedback from such code calculation may be necessary and must be carefully considered. With main reference to an integral test facility, an effort should be made to check whether the considered experiments have been undertaken in other facilities, i.e. similar tests or counterpart tests.

Phenomena expected to be relevant in the separate-effect test cannot be simulated in the concerned integral test facility due to scaling limitations and design compromises. In this case, it may be important to validate the code against a separate effect test, specifically concerned with that phenomenon. For example, the counter-current flow limiting (CCFL) phenomenon at the core' upper tie-plate usually is distorted in the integral-effect test due to the one-dimensional behaviour; in this case, a properly designed and operated separate-effect test should be used in validating the code.

Many of the boundary- and initial-conditions cannot be controlled by scaling criteria. In this case, compromises are necessary to reduce the impact of those on the test data. A few examples are the pump's characteristics, heat losses, pressure distribution, valve operation, and fuel simulators. Detailed descriptions can be found in the document, NEA/CSNI, 2001.

All scaling laws have certain advantages and disadvantages. Consequently, a facility generally cannot simulate all phases of a prototype behaviour, and some phases of the transient may be distorted, relative to the expected prototype behaviour. This problem is more pronounced in small scale- test facilities. The typicality of the experimental data obtained in scaled-down test facilities often is questioned due to inherent scaling distortions stemming constraints in its design and simulation, NEA/CSNI, 1996. To assure the adequacy of the experimental data obtained in scaled-down test facilities, it is beneficial to perform similar or counterpart tests under the same accident conditions with different scaling criteria and design concepts. The objectives of these tests are to evaluate the effect of scaling criteria and scaling distortions on the evolution of the transient. Also, it is beneficial to assure the code's scalability against a set of experimental data performed in different facilities with different scaling criteria and design concepts. Counterpart tests are useful for designing the criteria of scaled-down facilities, code validation, and the scale-up of the experimental data for real power plants, D'Auria et al., 1988, and Bovalini et al., 1992.

The distinction between similar and counterpart tests is the subject of disputable. Similar tests refer to some tests which, although they do not meet exactly the conditions of a counterpart test, have a great similarity in the initial and boundary conditions, NEA/CSNI, 1996. In this report, the general conditions are discussed for planning a counterpart test. As an example, the following parameters should be preserved for the counterpart tests:

Initial conditions

– Same thermodynamic state: Pressure and temperature in each section of the facility
– Same-scaled mass-flow rate
– Same velocity in the main components, if possible
– Same power-to-volume ratio

Boundary conditions

– Preservation of the sinks or sources, mass flow-rate over volume ratio
– Power- to volume-ratio
– Same actions based on actual signals

An early CSNI report, NEA/CSNI, 1987, deals with a wide list of separate and integral tests including similar and counterpart tests and was extensively updated by the report NEA/CSNI, 1996.

2.3.6 Fuel-rod ballooning

Clad ballooning and the resulting partial blockage of flow is one of the major concerns associated with the ability to cool partially blocked regions in a PWR fuel-assembly during a LOCA transient, Grandjean, 2007. Because of clad ballooning the fuel may fail during reflood phase. The characteristics of the clad ballooning and the flow blockage vary according to the LOCA scenarios, and the design of the fuel assembly. Several experimental programs were devoted to restore coolability after clad ballooning. The major experimental programs include FEBA (Ihle & Rust, 1984), SEFLEX (Ihle & Rust, 1987), THETIS, (Jowitt et al., 1984), Achilles (Pearson & Dore, 1991), CEGB, and FLECHT-SEASET (Loftus, 1982). In addition, several analytical researches were carried out in association with the experimental programs. Flow-blockage models in the COBRA-TF system's analysis code were developed and validated based on the FEBA- and FLECHT-SEASET-test results.

Clad ballooning occurs during the blowdown phase during a large break LOCA, Grandjean, 2005. Several in-pile tests such as those at ANL (Yan et al., 2014), and the Halden tests, NEA/CSNI, 2010, showed that fuel debris accumulated in the ballooned region, resulting from fuel fragments dropped from the upper regions of the core into the ballooned region. The burst of the clad and the possible relocation of fuel inside the ballooned regions appear around 800 °C. These relocations were initiated at the time of the cladding burst at the early stage of reflood during a LBLOCA. The fuel relocation causes a local power accumulation and a high thermal coupling between the clad and fuel debris in the ballooned regions. Thus, the fuel's relocation might affect the peak cladding temperature, the oxidation rate, the hydrogen uptake, and the quenching behaviours.

Recently, IRSN reviewed the experimental programs for the coolability of partially blocked core performed in the 1980s, Grandjean, 2008. The previous experiments did not consider the phenomena of fuel relocation and the resulting increase in local power in the ballooned regions. In addition, the previous experiments did not take into account the high thermal- coupling between the clad and the fuel. The estimated maximum blockage ratio was assumed to be about 71%, inferred from the NUREG-630 review. Thus, the previous experimental- and analytical- results are not considered to be conservative, Grandjean, 2008. In addition, several earlier experimental results imply that a flow blockage with high blockage ratio and long blockage length can entail a significant increase in the clad temperature in the ballooned regions, especially under low re-flood and low-pressure conditions. Therefore, the coolability of partially blocked core with fuel relocation is one of the important thermal-hydraulic safety issues in the revision of current LOCA acceptance criteria. The coolability in partially blocked core under medium- and high-pressure conditions is one of the unresolved issues in thermal-hydraulic safety.

The following phenomena affect the coolability of partially blocked core (Fig. 2-3):

- Flow redistributions between the ballooned regions and the bypass regions (non-ballooned regions).
- Break-up of droplets at the entrance of the blockage regions.
- Reduction of the coolant’s velocity and resulting fall of droplets down on the upper surface at the blockage’s outlet regions.
- Enhancement of single-phase heat transfer in the blockage regions.
- Reflood heat transfers due to the local power increase by fuel relocation.
- Azimuthal temperature gradient in the cladding combined with the anisotropy of the cladding material (effect known as Hot Side Straight Effect HSSE) tends to azimuthally localize the cladding deformation and consequently limit the overall deformation at rupture and the flow blockage.
- The occurrence of contact among rods affects the thermal heat transfer in the contact region.

No systematic scaling study has been performed on the coolability of partially blocked core. First, the geometries of flow blockages and fuel assembly should be preserved to assess their geometrical similarity. The linear power-rate should be preserved in a scaled-down test facility. The Reynolds number might have an influence on the single-phase heat transfer. For identifying a similarity in the droplets’ behaviour, the Weber numbers of the droplets should be preserved in a scaled-down test facility.

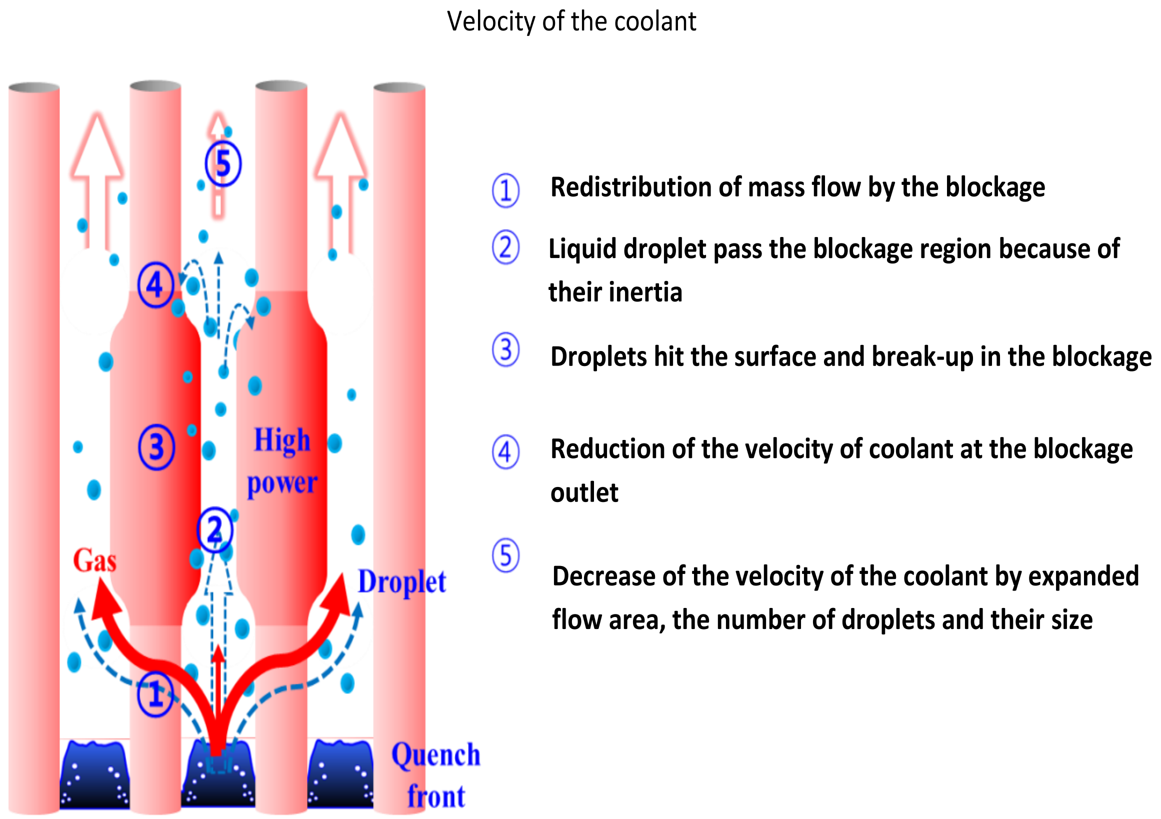


Fig. 2-3 – Major phenomena in the partially blocked core.

2.3.7 Special components

The reactor system contains many components that are important for its performance during normal and accident conditions, such as pumps, steam separators, advanced accumulator and jet pumps. Inappropriate scaling of these components may result in distortions in the reactor's condition, and thus requires evaluation.

The reactor's coolant pump is a good example to demonstrate the importance of scaling. It provides a pressure increase or head to overcome frictional losses in the system. The requirement for the pump is to assure a pressure rise to meet frictional losses in the design flow. The head flow curve for the pump for single-phase flow is governed by a specific speed, as defined below (symbols are defined in the list).

$$N_s = \omega\, q^{1/2} / H^{3/4} \tag{2-14}$$

The direction of flow in the pump changes from radial flow pump to axial flow pump as the specific speed increases. Pump can be designed to meet this condition at normal operation.

During depressurization transients, the system has a two-phase flow. When this flow reaches the pump, it undergoes separation in the pump channels and that leads to lower pump head. This represents a degradation of the pump's performance. It directly affects reactor safety as flow reversal in the core depends on the degradation of the head. The phase separation is less in axial flow pumps (higher specific speed) than in those with radial flow pumps.

Schneider & Winkler, 1987, suggested that the two-phase performance of the pumps depends on specific speed, void fraction, flow, pressure, and size. They provided some non-dimensional groups to account for the flow's stratification, slip in phases, and the effect of the flow coefficient on the degradation of the pump's head. An analytic study by Furuya, 1984, indicated that the degradation occurred due to increase in the liquid's velocity, the slip between the phases, and changes in the void fraction. It should be noted that this study did not include vapor compressibility, and that the increase in void fraction is due to relative velocity of the two phases.

The pump's performance is reported in terms of homologous curves. The parameters representing the performance are obtained from the non-dimensional analyses-Buckingham π Theorem. There are five independent variables, namely, pressure increase (ΔP), rotation (ω), flow rate (Q), diameter (D), and density (ρ) leading to two non-dimensional groups.

$$\pi_1 = \frac{\Delta \mathrm{P}}{\rho\omega^2 D^2} \tag{2-15}$$

$$\pi_2 = \frac{\omega \mathrm{D}^3}{Q} \tag{2-16}$$

$$H = \frac{\Delta \mathrm{P}}{\rho} \tag{2-17}$$

We can create alternate groups that can be transformed into homologous groups by eliminating the angular speed from π_1. In addition, the parameters can be normalized with design values so that group values remain within 1.0 and -1.0 as far as possible. The geometric parameter, D, will normalize to 1.0.

$$\frac{H^*}{Q^{*2}} = F\left(\frac{\omega^*}{Q^*}\right) \tag{2-18}$$

We can get another set from the π groups:

$$\frac{H^*}{\omega^{*2}} = F\left(\frac{Q^*}{\omega^{*2}}\right) \tag{2-19}$$

This provides a general format for presenting any pumps' performance curves based on non-dimensional analyses and normalization with design values.

Three possible phenomena in the pump's channels degrade its performance. These are the slip between the vapor and the liquid, phase separation, and the vapor's compressibility. The current model uses the inlet's void-fraction-based interpolating function between single phase and two-phase fully degraded (α > 0.9) performance. The phase separation also could depend on the flow field upstream of the pump. The study reported in CSAU indicated that the pump's size and specific speed impacts the degradation of its performance. Most models only account for the effect of the inlet void fraction on pump performance.

A CSAU study, USNRC, 1989, reviewed five sets of data from small-scale pumps to 1/3 sized Westinghouse pump (Table 2-1). The analyses are described in Appendix I of USNRC, 1989. It was shown that pumps with larger specific speed degraded less, a larger pump with same specific speed degraded less, and data collected at higher pressure degraded less. The AP1000 pump is a large pump with a specific speed of 6050 in units of gpm, ft, and rpm. As shown in Table 2-1, it will have larger specific speed than any other pump. Based on the CSAU analyses, it will have smaller effect from the void fraction on performance. Currently, the Westinghouse 1/3 scale is the largest pump data that is widely used. As the prototype pump (both the AP1000 and regular PWR) will degrade less than Westinghouse 1/3 scale pump the model based on Westinghouse Test data is expected to produce conservative results for LBLOCA because it will lead to stagnation and the reversal of flow sooner (i.e. with respect to what measured in the concerned experiments) during the blowdown phase of the LBLOCA.

There are other special components that are active during accident scenarios. Codes have correlations based on large-size tests from the vendors.

Recently, MHI provided a model for advanced US PWR. The advanced accumulator has a vortex device at the bottom of the tank that replaces a valve in surge line and control the flow rate. Initially there is flow through central pipe into vortex chamber leading to radial flow in the vortex chamber exiting in surge line. After certain amount of flow has been released, there is no flow through the central pipe and only tangential flow at the bottom of the tank into the vortex chamber. This leads to vortex flow, an increase in friction, and a reduction in flow. MHI who had a full length, 1/2 sized test facility developed a model to connect pressure difference between top of the accumulator and cold leg, liquid level for early flow and later vortex flow case. As the vortex chamber was only half size, there was some scale distortion that was estimated using CFD analyses of the prototype and the test facility. Scale distortion was very small (few percentages). However, there was controversy in the procedure of CFD analysis which was not consistent with the best practice guideline and ASME V & V 20 that contributed additional biases.

Table 2-1 – Examples of centrifugal pumps.

Note: A/W is air/water mixture, S/W is steam/water mixture

	Westinghouse			Bingham-Williamette Primary Coolant Pump		Byron-Jackson Primary Coolant Pump			RS111
Parameter	**PWR**	**AP 1000**	**W. Small Pump**	**PWR**	**B&W Pump**	**PWR Primary Coolant Pump**	**C-E Pump**	**Creare Pump**	**KWU Pump**
Scale	1/1	1/1	1/3	1/1	1/3	1/1	1/3	1/20	1/5, RS111
Rated Volumetric Flow Rate (gpm)	94600	78750	6210	104200	11200	87000	3500	181 (219)	3148
Rated Total Head (ft)	290	365	64.4	397	390	252	252	252	293.7
Rated Speed (rpm)	1190	1800	1500	1190	3580	900	4500	18000	8480
Specific Speed rpm (gpm)0.5/[(ft)0.75 s]	3200	6050	5190	4319	4317	4200	4200	4200	6700
Fluid	S/W	S/W	A/W & S/W	S/W	A/W	S/W	S/W	A/W & S/W	S/W
Pressure (psia)	15-2250	2500	15-420	15-2250	20-120	15-2250	15-1250	A/W at 90 S/W at 400	435-1305

2.3.8 Local phenomena in the core at sub-channel level

The flow in a LWR mainly is an axial flow quasi-parallel to the rods in most situations of interest. However, some deviations from the pure axial flow exist due to unequal pressure losses in adjacent sub-channels, or adjacent assemblies. Cross-flows are created to recover horizontal pressure equalization. Also, in axial- or quasi-axial-flows there are radial transfers of mass, momentum, and energy which are due to three types of mechanisms, e.g. Drouin et al., 2010, Chandesris et al., 2006, Bestion, 2014, Bestion, 2015, and Bestion & Matteo, 2015:

1. Molecular diffusion of momentum and heat.
2. Turbulent diffusion of mass momentum and heat associated to time fluctuations of the flow's variables, such as velocity, temperature, and void fraction.
3. Dispersion of mass momentum and energy due to non-homogeneity of flow variables, such as velocity, temperature and void fraction.

In a 3-D model approach for a porous medium, a double time and space averaging of local instantaneous equations is needed, with a time averaging that filters turbulent fluctuations, and a space averaging that includes at least one sub-channel (sub-channel analysis) or several sub-channels or assemblies. Time-averaging of non-linear advection terms is known to produce averaged advection terms and turbulent stresses (momentum equations), or energy fluxes (energy equations) in a RANS approach. Space averaging of non-linear advection terms generates additional dispersion terms.

Crossflows and radial-transfer terms may be either dominant phenomena or negligible ones, depending on the situation of interest.

In a CHF investigation, neighboring sub-channels may receive a different level of power, and the occurrence of CHF depends primarily on local flow conditions that are very sensitive to dispersion terms and crossflows. In such conditions, adequate experiments are built with full-power full-pressure scale rod-bundles and with radial power profile to investigate CHF conditions.

Sub-channel analysis codes are validated on these data, and the CHF is correlated as function of the sub-channel-flow parameters. This requires that the sub-channel analysis code models radial diffusion and dispersion, and is validated with specific tests that measure radial mixing and cross-flows.

In other situations, such as LOCAs, radial transfers may play a minor role, or may be easily treated:

– In a core uncovery during a SBLOCA, the top part of the core may be cooled by pure steam and radial transfers of heat between sub-channels or between assemblies may be neglected compared to wall heat flux. This explains why most SETs and IETs used to investigate SBLOCA even do not represent the radial power profiles that may create radial transfers. This explains also why current SYS TH codes do not model diffusion and dispersion, and can predict correctly core uncovery, even with radial power-profile, Morel & Bestion, 1999. However, a modelling and quantification of these effects could improve the accuracy of predicting the maximum cladding temperature.
– In a core uncovery during a SBLOCA, or in the rewetted zone of a core during a LBLOCA-reflooding process, the radial power profile creates strong radial mixing due to gravity- driven cross-flows. High-power assemblies create more steam than low power assemblies, and gravity tends to homogenize the void fraction between assemblies. In such cases, SYS-TH codes with 3-D capabilities of with cross-flow junctions can predict the cross-flow that has a larger effect than diffusion-dispersion, Chandesris et al., 2013. This explains why current SYS TH codes, which do not model diffusion and dispersion, can predict reflooding tests both with and without a radial power-profile, Morel & Boudier, 1999, and Dor & Germain, 2011. However, modelling and quantifying these effects could improve the accuracy of predicting the maximum cladding temperature.

2.4 Addressing scaling issues

As mentioned earlier, scaling has been widely applied in researches in science and technology. It also is well acknowledged that a complete similitude is impossible, and scaling distortions are inevitable. In nuclear-reactor applications, the distortions resulting from scaling are related to nuclear safety because the design, operation, and analysis of nuclear reactors are tightly related to scaling. Therefore, resolving scaling issues is an important step toward nuclear safety. Regulatory agencies acknowledge this fact by including the scaling evaluations in standard regulatory procedures. Their objective is to ensure the validity of the tools used in safety analyses, and to address the scaling distortions in the reactor's design and operation. Two regulatory procedures that involve scaling are introduced here. In the first procedure scaling evaluation is used in guiding the model development and assessment. In the second procedure, scaling distortion is one of the uncertainties that must be quantified.

2.4.1 Evaluation models development and assessment

To demonstrate the safety of a nuclear-reactor design, the licensees are required to present plant- specific safety analyses for evaluation. Before the safety analyses are presented, the tools used in the safety analyses must be reviewed and accepted. In regulatory world, the tools commonly are referred to as "evaluation models (EM)", USNRC, 2005. According to NRC's definition, EM is a calculation framework for evaluating the behaviour of the reactor system during postulated Chapter 15 events, USNRC, 2000, which include one or more computer programs, and all other information needed for use in the target application.

Most regulatory agencies worldwide have their own regulatory procedures regarding EM development and assessment. For example, in December 2005, the USNRC published Regulatory Guide 1.203 – Evaluation Models Development and Assessment Procedure (EMDAP), USNRC, 2005, as also discussed in Chapter 1. This regulatory guide is intended to provide guidance for developing and assessing EMs for accident- and transient-analyses. EMs developed under this guidance will sufficiently provide a reliable framework for risk-informed regulation, and a basis for estimating the uncertainty in understanding transient- and accident-behaviours.

EMDAP is a multiple-step procedure, as shown in Fig. 2-4. Most steps in the procedure are irrelevant to this project. However, in Element 2 – Develop Assessment Base shown in Fig. 2-5 – scaling plays an important role. Licensees are expected to provide a scaling analysis, and to identify similarity criteria in Step 6. And, in Step 8, licensees are expected to evaluate effects of IET distortion, and the capability of scaling up SETs.

The objective of scaling analysis is to demonstrate that the experimental database and the developed model based on the database are applicable to full-scale plant transient analysis. It is impossible to attain complete similarity between the full-scale plant system and the scaled experiment. Therefore, scaling analysis is performed to show that the collective behaviour of the experimental database is sufficient to represent the expected response in the postulated plant transient. The analysis also is for investigating if the models and the code represent important phenomena in the plant transient.

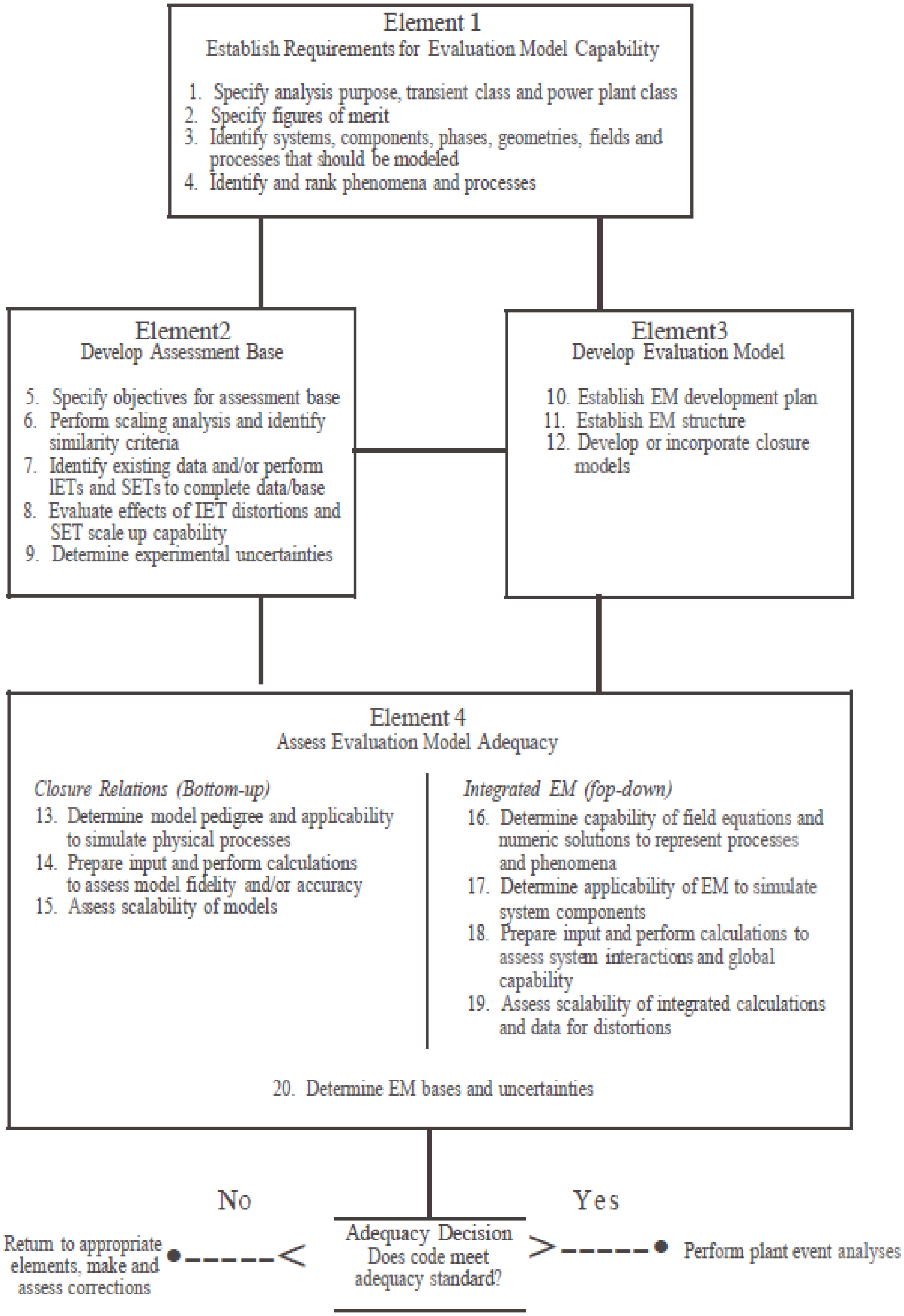

Element 1
Establish Requirements for Evaluation Model Capability
1. Specify analysis purpose, transient class and power plant class
2. Specify figures of merit
3. Identify systems, components, phases, geometries, fields and processes that should be modeled
4. Identify and rank phenomena and processes
Element2
Develop Assessment Base
5. Specify objectives for assessment base
6. Perform scaling analysis and identify similarity criteria
7. Identify existing data and/or perform IETs and SETs to complete data/base
8. Evaluate effects of IET distortions and SET scale up capability
9. Determine experimental uncertainties
Element3
Develop Evaluation Model
10. Establish EM development plan
11. Establish EM structure
12. Develop or incorporate closure models
Element 4
Assess Evaluation Model Adequacy
Closure Relations (Bottom-up)
13. Determine model pedigree and applicability to simulate physical processes
14. Prepare input and perform calculations to assess model fidelity and/or accuracy
15. Assess scalability of models
Integrated EM (top-down)
16. Determine capability of field equations and numeric solutions to represent processes and phenomena
17. Determine applicability of EM to simulate system components
18. Prepare input and perform calculations to assess system interactions and global capability
19. Assess scalability of integrated calculations and data for distortions
20. Determine EM bases and uncertainties
No
Yes
Return to appropriate elements, make and assess corrections
Adequacy Decision
Does code meet adequacy standard?
Perform plant event analyses

Fig. 2-4 – The EMDAP process (USNRC 2005).

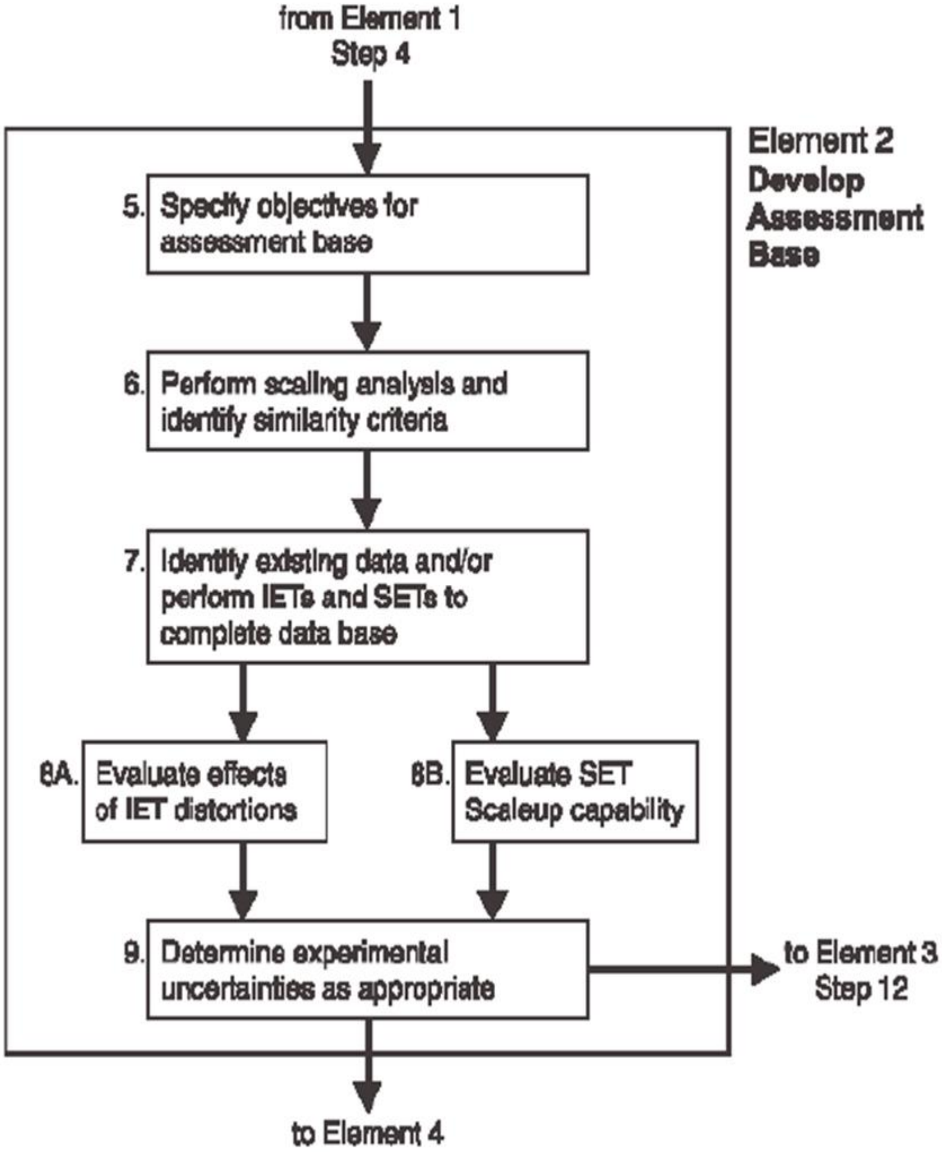


Fig. 2-5 – Scaling in the EMDAP process (USNRC 2005).

The scaling analysis usually consists of two parts – a top-down integral response analysis, and a bottom-up local phenomena analysis, Zuber et al., 1990, and Zuber et al., 2007. The top-down analysis usually includes deriving non-dimensional parameters that govern the similitude between the plant and the test facilities. It checks whether the experiment results can be scaled-up well using these non-dimensional parameters and verifies that their range covers the plant's conditions. In the bottom-up analysis, some important localized phenomena identified in the PIRT process are studied. Empirical formulas are employed to verify if the geometry and fluid conditions of the test facilities and the plant satisfy them. Differences are explored to explain the facility differences and to infer the behaviour of the full-size plant.

Distortion could arise from many factors in the IET's design and operation. The scaling distortion from IET usually is due to missing or the compromised scaling of important phenomena, along with the initial/boundary conditions in scaled facilities. These distortions should be evaluated and quantified for determining acceptance. The correlations to be used in EM should be developed in SETs at various scales. The scalability of these SETs also needs to be evaluated to determine the acceptability of using the correlations in full-scale plant simulations. CSAU methodology, USNRC, 1989, describes the rationale and techniques associated with evaluating the scale-up capabilities of computer codes, and their supporting experimental databases. In the Step 10 of the procedure of CSAU methodology, the quality and evaluation of the model and the correlation documents, along with the assessment report of the code, are used to identify and evaluate closure correlations and their capability to scale-up the important process listed in the PIRT for the intended scenario. In the meantime, the effects of distortions in some phases of the process are evaluated for bias specification if they are deemed important. The detailed steps and an example are captured in the Appendix C of the CSAU methodology, USNRC, 1989.

2.4.2 Requirements of CSAU

In 1988, the USNRC revised the ECCS rule that allowed the use of best estimate code in performing safety analyses with the uncertainty accounted for, USNRC, 1988, and USNRC, 1989. This alternative approach commonly is referred to as the best estimate plus uncertainty (BEPU) method that opened a new era of safety analysis. In this approach, the premise is to account for the uncertainty in the results of the analysis. To illustrate this approach, USNRC also issued the Code Scaling Applicability and Uncertainty (CSAU) methodology, along with a real NPP example to demonstrate the uncertainty quantification using the TRAC code. This methodology received favorable responses in the community, particularly from the Advisory Committee on Reactor Safeguard (ACRS) of the USNRC – "The CSAU methodology provides a technical basis for uncertainty quantification within the context of the revised ECCS rule and confirms the worth of using best-estimate codes to license and regulate reactors. The use of best-estimate codes and uncertainty quantification provides the basis for reductions of ECCS surveillance requirements, increased operating power, extensions of reload cycle times, reduction of steam generator tube plugging constraints, etc.", USNRC, 1989, and USNRC, 1989a.

Code Scaling, Applicability and Uncertainty (CSAU), USNRC, 1989, evaluation methodology was developed by the USNRC to provide a systematic, auditable method of estimating uncertainty in the prediction of safety parameters from best estimate computer codes. As the methodology relied heavily on tests of separate effects and integral effects, recognition of scaling distortion of these tests and their contributions to the overall uncertainty, is an important aspect of CSAU.

CSAU is a 14-steps procedure. It starts from identifying scenario or transient, power plant and important phenomena based on Phenomena Identification and Ranking Table (PIRT). Then the code for safety analysis is chosen for the transient. Based on the code manuals, the code applicability is determined and an assessment matrix is established. Nuclear power plant nodalization is established and safety calculation is performed. Based on comparing the calculation results with SET (separate effect test) and IET (integral effect test) data set, the nodalization is modified and iterated. Then, the overall uncertainty of the calculation result is evaluated. In the CSAU procedure, three uncertainty sources are quantified – the code and experiment accuracy (Step 9), the effect of scaling (Step 10) and the reactor input parameters and state (Step 11). CSAU ends with the estimation of total uncertainty. Fig. 2-6 shows the methodology flowchart.

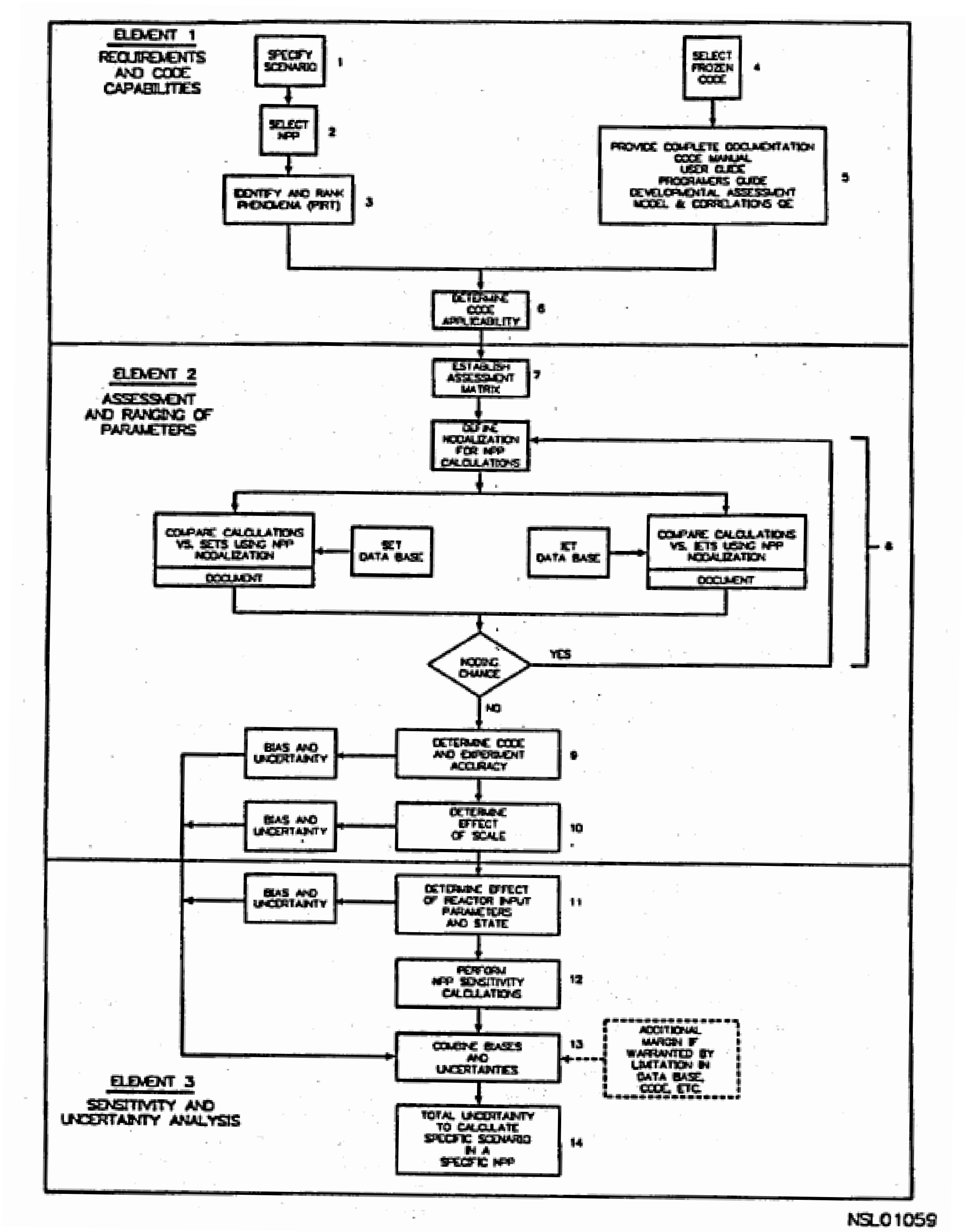


Fig. 2-6 – CSAU Methodology Flow-chart.

The code applicability is assessed in the Step 6 of the CSAU. The PIRT identifies the important phenomena for the given scenario and plant. The code's formulation, model, and correlations are reviewed to assess if the code has the model/correlations to simulate the important phenomena, and if the code has scale-up capability. It is recognized that while the formulation may be general, the correlations or constitutive relationships are empirical. The code's scalability will depend on these correlations and the underlying tests. If the tests scale the plant for the phenomena of interest, then correlation is applicable. However, if correlation was derived from tests that do not scale the plant, the code may not scale up the plant. Code validation with scaled tests or counterpart tests is another way of assessing code's ability to scale.

The Step 9 is about estimating the code's uncertainty based on experiments, and Step 10 is about determining the effect of scaling and the resulting bias and uncertainty. Normally, these two steps are combined. In them, the uncertainties of code models that are estimated and are input to Step 13 were all the uncertainties and biases are combined. Details related to the scaling are described in Section 4.4.2.1, Scaling in CSAU.

As an example, CSAU was applied to a PWR's LBLOCA. The PIRT development in Step 3 identified five important phenomena; break flow, stored energy, and pump's performance under two-phase conditions, steam binding, and the ECC bypass. Some of these phenomena were further broken into sub-phenomena. These are described here to illustrate the evaluation process.

Uncertainty in break flow prediction was estimated for subcooled and saturated conditions as the code has two different models. Marviken tests, Marviken Project, 1974, provided full-scale data, and so the uncertainty estimate did not need any additional scaling bias. The ranging of the parameter (surrogate for the critical flow model) and bias were estimated for subcooled- and saturated-choking.

The review of available counterpart tests for primary pump showed scale-dependencies for the two-phase performance or degradation of the head. The two-phase flow has its own characteristic length, such as bubble size, and the larger the size of the flow channel, the smaller is the effect on the degradation of performance. In addition, specific speed also has an effect on degradation. Larger specific speed-pumps have more axial flow and tend to degrade less. The largest size pump in the database is the Westinghouse pump at a 1/3 scale. The specific speed is around 5200 and is the same as Westinghouse PWR pump. Comparing two-phase performances of the pumps at different scale, it was concluded that larger pumps experience smaller degradations. Therefore, Westinghouse's 1/3 scale data can be used for the NPP pump. It will have a slightly larger degradation than the NPP pump. However, this will be conservative as pump flow will decrease sooner at least for LBLOCA.

The stored energy in the fuel was rated very high in the PIRT. It was shown, through sensitivity analyses, that the stored energy is the most sensitive to four parameters, namely, gap conductance, peaking factor, fuel's thermal conductivity, and convective heat transfer. There are no scaling issues with the first three parameters since they are either independent of the length scale (thermal conductivity) or boundary conditions (peaking factor) or no scaled study is available (gap conductance). However, the convective heat transfer was further broken into two parameters, minimum film-boiling temperature, and the convective-heat transfer coefficient. It is recognized that a lower Tmin will lead to an earlier transition to film boiling in the blowdown phase and delay the transitioning to subcooled boiling in the reflood phase, and leading to higher peak-clad temperatures. There were data from a large number of tests both for the tubes and the rods, and for a large range in pressure. The TRAC models were compared with the available data. It was recommended that a homogenous nucleation temperature be used as it gave a lower Tmin than that predicted by TRAC. The spread of all the data was 360 °F and half of this value (180 °F) was recommended for the range. Here, the approach is to use a conservative mean value and the range. The other contributor to heat transfer is the heat-transfer coefficient. During the process of comparing the data it was found that TRAC over-predicted heat transfer coefficient at void fraction greater than 0.75 and that the cause of this over prediction is the film boiling heat transfer coefficient. In the actual code application, it is recommended that the Forslund-Rohsenow film-boiling correlation is removed, and the uncertainty in the remaining heat-transfer models is accounted for through a multiplier with a range of 0.75 to 1.25. The scaling effect was included in the overall uncertainty in the heat-transfer model.

Another important phenomenon that affects the PCT is steam binding in the hot leg and the steam generator's u-tube. The liquid carried in the steam generator vaporizes and creates a high pressure in the upper plenum, delaying the onset of re-flooding, leading to a higher PCT. The sub-phenomena that control vapor-binding are the drops coming from the liquid interface in the core, and the entrainment of these drops into the hot leg and steam generator. Modelling of two Slab Core Test Facility (1/21 size) tests with TRAC indicated that the code under-predicts the entrainment flow to the hot leg. A set of multipliers were developed to apply to TRAC models to increase the number of drops emanating from the interface, and also to increase interfacial shear-stress to match the entrainment to the hot leg and the steam generator. The SCTF is a small facility. It is assumed that as correlations that are being affected are localized; they are not dependent on the size of the facility.

The phenomenon of Emergency Core Coolant (ECC) bypass is the most important phenomenon as it determines the refill phase and the flooding and quenching of the core. It affects the reflood PCT. The ECC injection is from the accumulator to the cold leg and to the downcomer. The steam up flow in the downcomer towards a break impedes the ECC flow through interfacial shear, thus delaying the reflooding phase. The other phenomenon that assists in refilling is the condensation of steam in the downcomer. When non-condensable gases are present in the downcomer, condensation is reduced and the ECC bypass is increased. The filling time of the lower plenum will depend on the combination of these phenomena. Many tests for ECC bypass phenomenon are available at different scales, such as CREARE (1/5), Battelle Columbus Laboratory (BCL) (2/15), and CREARE (1/15) A comparison of the predicted and measured lower plenum filling rates indicated that TRAC over-predicted the filling rates for small scale facilities, and under-predicted them for a 1/5 scale facility. Data from a full-scale facility, UPTF, also became available. The TRAC code under-predicted the filling rate for UPTF. As UPTF is full-scale test, the model's uncertainty and the corresponding bias in PCT was estimated directly. In addition, the TRAC code did not have a model for non-condensable gases. A separate analysis was performed to estimate the amount of non-condensable gas coming out of the coolant, and its impact on delaying reflood phase and the corresponding bias in reflood PCT.

CSAU also reviewed many integral test facilities at different scales, such as Semiscale, LOFT, CCTF, SCTF, and PKL. It was concluded that all the blowdown PCTs were within the 95% tolerance lines when plotted as function of the linear heating-rate. This finding indicated that this is most dominant phenomenon in the core. However, there are other system-effects such as pump break, that determine the time at which the heat transfer has degraded enough so that temperature rise is only function of decay heat. In the case of reflood peak, the plots were made with the rise in core temperature as function of the reflooding rate. It indicated that cores were well scaled in different facilities varying in size from Semiscale (1/1700) to CCTF (1/21). However, the reflooding rate will be governed by system performance and any scaling distortion there will affect the rate of reflooding. The study showed, as expected, that temperature rise declines with an increase in the reflooding rate.

Facilities scale the blowdown-phase well. However, the scaling effect becomes more pronounced for the refill- and reflood-phase due to the interaction of many scaled phenomena, such as the ECC bypass. To find scale similarities the PCT were plotted as function of flow in the core. PCTs from different facilities correlated well. As the core thermal-hydraulic was decoupled from rest of the system in the refill/reflood phase in this study, the plot suggested that core was scaled well in the all the facilities.

The CSAU demonstration considered the important phenomena to the fullest extent in the 1980s. However, some phenomena were not considered and treated in the uncertainty estimation. One phenomenon is the radiation heat transfer. As the rods are uncovered in the LOCA process, the transfer of radiation heat increases due to the large difference in temperature between the cladding and internal structure, or between the covered- and uncovered-rods. Another uncertainty is the use of electrical-heating rods in the core simulator in test facilities. The rod's outer diameter may be the same as the actual fuel rods; however, the internal composition of the rod and its geometry may differ greatly between the facilities. Since the stored energy is related to the properties of the rods' internal materials, their use may cause a large deviation in the scaled tests. Another important phenomenon is the ballooning of the rod when its temperature rises to a certain level after being uncovered. Flow geometry in the core may be changed in this case. The details are described in Section 2.3.6.

2.5 Key findings

In this section, a quick review is made to introduce subjects related to scaling. It is well agreed that scaling is an indispensable for verifying the safety of nuclear reactors. However, the scaling distortions revealed in the past few decades indicated additional research and developments are needed to perfect this technology. The review findings are summarized below.

1. In reviewing scaling achievements, an important issue was identified that either the experiments or the computational tool could inaccurately predict the essential phenomena, e.g. the flashing in the SEMISCALE downcomer or in the SG generator downcomer in the LOBI experiment. This was an achievement in the sense that we clarified the facts and identified the deficiency of tools. It also was a warning that new challenges in thermal hydraulics may remain undisclosed, particularly in the new reactor-designs.

2. In reviewing the scaling methods and evaluation models, the group finds that the methods and evaluation tools both have limitations that cause distortions and uncertainties. There are no perfect tools or methods available to cover the range of nuclear applications. Therefore, the choice of scaling a design for intended application is a compromise of scaling distortion.

3. As the reactor's design advances, the quantification of scaling distortion becomes increasingly important. It is essential to relate the quantified scaling distortion to the figure of merit. In terms of scaling limitation, the quantification of distortion is an important step in making this determination. As to the propagation of scaling distortion, a method to evaluate the accumulated distortion as a function of time also is in need in a phased transient.

4. In nuclear thermal hydraulics, many key complex phenomena remain unresolved in terms of scaling. Some phenomena lack creditable experimental data to validate correlations that might may serve as starting points for scaling design, e.g. two-phase choke flow, and counter-current flow limitation. Some other phenomena are still in early stage in acquiring detailed essential experimental data to facilitate correlations, e.g. entrainment and de-entrainment. Other complex phenomena, e.g. fuel ballooning simply are too complex to develop a scaling method. The difficulty raised by these examples indicates additional research and development are needed in the scaling sciences.

5. In safety analysis, scaling is an important source of uncertainty. The evaluation model contains numerous experimental correlations wherein scaling distortions are embedded. The nodalization also could also include scaling effects that influence the results in the reactor simulation. Therefore, the applicability and the scalability are two main concerns in the model. Currently available approaches to meet safety requirements are focused on these two areas.

3. SCALING TECHNIQUES

3.0 Introduction

The confirmation of safety in reactor design and operation is done by estimating the thermal-hydraulic response of the prototype in reference reactor, using the data from experiments, and/or computer code calculations. Since it is difficult to use the reference reactor to obtain all the necessary data, especially for the thermal-hydraulic response during postulated accidents, it is inevitable that we must use simulation experiments with scaled test-facility and computer-code calculations with some appropriate scaling techniques. Computer codes have the capability to extrapolate knowledge from scaled experiments to the prototype. However, uncertainty in the predicted results should be estimated by using the existing experiment data afterwards. In this sense, the role of the experiments used to develop and validate computer codes is very important until the final stage, thus the validation of the calculated results.

Well before we obtain good computer methods, both of the computer and codes, the expected local phenomena were estimated by using a dimensional analysis, such as Buckingham Pi theorem. Since the Karman vortex may appear behind a high mountain or an island, from satellite observations, similarly to those observed behind small objects, one may believe that the extrapolation of phenomena should be sound up to the prototype from small-scale experiments, provided that the Reynolds number lies within a certain range, for instance. However, as experienced, thermal-hydraulic phenomena may not always be linearly extrapolated nor even interpolated, depending on such parameters as the fluid physical properties, the complexity of channel geometry, and 3-D steam-water two-phase flows with dynamic change of flow conditions. The complexity significantly increases when dealing with the transient phenomena in the reactor systems.

When the models and correlations are prepared, separate-effect tests usually are conducted under steady-state conditions with well-defined initial- and boundary-conditions within the scaled test facility designed by using some appropriate scaling method. These are also important when data measurement is done under stable conditions, while unstable conditions usually exist. Developed models and correlations should cover the test conditions; however, they may not cover areas outside them, in the domain of space (size, geometry), physical properties and time.

A few examples associated with the shortcomings in experiments are as follows, as already discussed in Chapter 2:

(1) Models and correlations including closure relationships (phase change, flow regime transitions., etc.) are developed using separate-effect tests (SETs) under steady, stable conditions (i.e. D'Auria et al., 1998, dealing with the Steady-State and Fully-Developed flow paradox), to firmly and accurately measure the data, whilst a safety analysis in general deals with unstable, non-equilibrium fast transients of 3-D multi-phase flows under interactions among reactor components, from full pressure to low pressure, down to atmospheric pressure. An integral-effect test (IET) then is employed to obtain the transient thermal-hydraulic response. Data are obtained within the range of conditions of the scaled facility that may not fully cover the prototype conditions. Scaling distortion is inevitable in experiments depending on the design of either SETs or IETs.

(2) CCFL phenomena in a PWR downcomer during the LB-LOCA refill phase are under the influences of the size of the downcomer gap and the 3-D gas-phase flow conditions, such as orientation and distribution/flow profile, Glaeser & Karwat, 1993, and Wolfert, 2008. This also is discussed in Chapters 2 and 4. Extrapolation from small-sized tests failed to properly explain the observed phenomena under the prototype size and conditions (e.g. see UPTF).

(3) The flow-regime transitions in horizontal pipe, such as wavy-to-slug and slug-to-wavy-dispersed (annular) are influenced by the pipe diameter and by the fluid and flow physical properties and conditions, including pressure, Nakamura, 1996. The density-modified (non-dimensional) Froude Number, viz., the liquid superficial velocity, for flow regime transition from wavy to the slug flow increases with pipe diameter. Furthermore, the slug flow may disappear when the pressure is higher than around 9 MPa because of the influences of the changes in physical properties of water and steam. However, there are no relevant data under the combined conditions of large diameter and high pressure, which is typical of the reactor.

(4) Measurement of the BWR feedwater flow rate by the nozzle test under prototypical conditions, Furuichi et al., 2014, concerning the extrapolation of high-Re condition (up to the order of 10^6 with conventional tests) to the BWR prototypical 'extra-high' Re condition (Re value up to 10^7 with a correspondingly large test). The ASME correlation used by Furuichi et al., 2014, for extrapolation may need special care, including the method for measuring pressure in experiments.

The predictability of computer codes therefore are validated against integral data from test facilities to confirm their applicability to the transient within the complicated geometry of reactor system, which, in most cases, is outside the range of conditions for separate-effect tests used to develop the models and correlations for the computer codes of concern. The capability of extrapolation/interpolation from the scaled facility, either of SETFs and ITFs, to the prototype then is confirmed provided that a good amount of experimental data covers the prototypic conditions that are required for safety analyses.

As above, the experiment itself takes the key role in the reactor safety in relation to the computer code. This also means that scaling techniques used to design the test facility should be the key to understand the validity of experimental data; thus, the results calculated by computer codes being developed by assembling models and correlations from experiments need to be validated by using many types of experimental data.

In Chapter 3, scaling techniques are described. The scaling process is started from scaling approach to develop a plan to design test facility and to understand obtained data; however, in this chapter, major scaling methods are first outlined to address local phenomena and integral response of reactor system. The scaling approach is considered after the description of the scaling methods. The experiments themselves then are discussed for the separate-effect tests (SETs) and the integral-effect tests (IETs) for such reactor types as PWR, BWR, VVER, and new LWRs as well as the containment vessel, especially on how scaling was considered in their design. Finally, as an approach to scaling from experiments, the roles and significances of counterpart-tests and similar tests are described, with some discussions on experiences from tests in daughter facilities.

3.1 Scaling methods

In this Section, all the major scaling methodologies are described, including those for local- and system level-phenomena as well as for designing facilities and corresponding experimental conditions. Here, the scaling approach is a kind of strategy to assure appropriate methods are applied.

Scaling methods then open the way to the derivation and application of experimental data, e.g. Sections 3.2 and 3.3 (in particular, Section 3.2.2.2). The discussion in these Sections shows the important role of SYS TH codes, which is an entrance to the discussion in Chapter 4. A suitable (and rigorous) V & V process is needed before applying the SYS TH codes and connected model (e.g. nodalization) to the NRS, and is necessary for addressing the drawbacks for scaling methods that were introduced in Section 2.2.2.

Due to inherent difficulties in conducting full-scale tests, most of separate effects tests (SETs), and the integral effect tests (IETs) have been performed in scaled-down test facilities with the assumption that the experimental results obtained are applicable and relevant to the full-scale reference reactors. The test facilities, as well as the operating conditions, should be properly scaled, and scaling distortions should be minimized so that they do not affect important phenomena and system global behaviour. The computer codes have been validated based on experimental data mostly generated from the scaled-down test facilities. The validated computer codes should be able to predict scale-up thermal-hydraulic phenomena and processes that may appear in the full- scale nuclear-power plants where the uncertainty of the predicted phenomena and processes can be evaluated accurately. For these reasons, the scaling plays key roles in the design and operation of experiments, and the validation of the computer codes, Boyack et al., 1989, and Zuber et al., 1990.

A major aim of the SETs is to investigate fundamental phenomena and processes at a local level and to prepare physical models and correlations for computer codes. On the other hand, IETs have been carried out to investigate the integrated system responses during reactor operation and accidents. To assure the proper simulation and preservation of important phenomena and processes at local level, or at system level, appropriate scaling methods should be established to scale-down those phenomena and processes from the prototype to the test facility. Scaling methods may affect the interpolation or extrapolation of the obtained experimental data within a certain range of applicability that may differ among the phenomena. Over the past years, a variety of studies were completed on scaling methods to establish similarity relationships between the prototype and the scaled-down test facility. A lot of scaling methods have then been established, which can be applied to both SETFs and ITFs. Depending on the local and/or system-wide phenomena of interest, different scaling parameters may be required to properly design and operate scaled-down facilities to either represent or simulate the phenomena expected to occur in the prototype.

In general, the scaling parameters for a certain phenomenon can be derived by applying a dimensional analysis (the empirical approach), or by dimensionless governing equations (the mechanistic approach). Dimensional analysis, such as Buckingham Pi theorem, can be adopted for scaling local phenomena by considering conventional non-dimensional parameters. The Buckingham Pi theorem provides functional relationships among the variables that govern the phenomena of interest. However, this theorem has inherent difficulties, such as the identification of important phenomena.

Another method to obtain the dimensionless parameters is by non-dimesionalizing the governing equations. However, the governing equations need a process of approximation and assumption(s) for thermal-hydraulic behaviour, especially in the prototype, because of the complexity in the representation of three-dimensional single- and two-phase flows. Related to this process, the following paradox may arise; balance equations, including constitutive terms that are, in principle, difficult to be solved/qualified because they do not undergo V & V processes, e.g. SYS TH codes, are used for deriving non-dimensional parameters and even performing scoping calculations within the framework of scaling analyses.

Buckingham Pi theorem and non-dimensionalization of governing equations are usually employed to derive the scaling parameters for many local phenomena. Scaling of the SETF then is mostly related to these methods to find the scaling parameters.

For the ITFs, it is necessary to preserve the important local thermal-hydraulic phenomena/processes, as well as the system behaviour. The geometric-, kinematic-, and dynamic- (chronological) similarity of thermal-hydraulic phenomena/processes should be preserved in the ITFs. First, global system behaviours, such as the natural circulation of single- and two-phase flows are preserved by using global scaling criteria that are obtained from non-dimensional governing equations. Second, local scaling criteria are obtained to preserve the important thermal-hydraulic phenomena that may happen in a component. Scaling distortions may occur in the simulated local phenomena because of the difficulty in matching the local scaling criteria with the global scaling ones, and the dearth of knowledge on the local phenomenon itself, Ishii et al., 1998. In scaling the local phenomena, identification and ranking (PIRT) play a valuable role for identifying important local phenomena and processes that should be preserved in the scaled-down facilities.

All the major scaling methods for the ITFs are described together with their major characteristics, merits, limitations, and their areas of application. The characteristics of the choices of height scale, time scale, and pressure scale are presented in relation to the scaling and design of ITFs, and thus, with the scaling approach.

In the sub-section on scaling approach, the characteristics and practical selections of the height scale, the time scale, and the pressure scale are described. The merits and limitations of the full-height and reduced-height facilities are presented. The characteristics of time preservation and reduction also are described. Lastly, three kinds of pressure scaling are described. It is general practice to perform scoping calculations using a system analysis code after the major scale ratios are determined and the basic design of a scaled-down facility is completed. The system-analysis code models are created for both the prototype and the scaled-down facility. The main objective of the scoping calculations is to investigate the similarity of a transient system behaviour. The analysis of the system transient behaviours including comparison among results from different calculations, contributes to identify the scaling distortions in the facility and to minimize the scaling distortions by optimizing the facility design.

3.1.1 Local phenomena

In general, the scaling parameters of local phenomena can be deduced from either the dimensional analysis (empirical approach) or the dimensionless governing equations (mechanistic approach). When the governing equations are unknown, then a simple dimensional analysis is useful to drive the important scaling parameters for any local phenomena. Dimensional analysis, such as Buckingham Pi theorem, identifies scaling parameters for a given local phenomena, and correlates the experimental data, White, 2001, and Kreith et al., 1999. Functional relationships among the scaling parameters can be determined experimentally to fully characterize a given local phenomena. The dimensional analysis can be performed without any knowledge of the governing equations and the nature of the phenomena. Dimensional analysis reduces the number of dimensional variables into a small set of dimensionless groups that facilitate understanding of the physical phenomena. The dimensional analysis starts by listing all the dimensional variables that are known to affect the phenomenon of interest. Selection of the important variables requires considerable experience and good knowledge of the nature of the given local phenomena. An incomplete set of dimensional variables results in inadequate dimensionless parameters. On the other hand the selection of too many dimensional variables results in too many dimensionless groups that complicate the analysis of the local phenomena.

There are many examples of dimensionless analysis, using Buckingham Pi theorem. For example, by applying Buckingham Pi theorem to an incompressible flow with constant viscosity, we can obtain dimensionless scaling parameters, such as the Reynolds number, Euler number, Froude number, and the Weber number. A single-phase fully developed forced-convection heat transfer can be well correlated by the Nusselt number, Reynolds number, and the Prandtl number that are obtained by dimensionless analysis. The similarity parameters for fluid-to-fluid modelling of the critical heat flux can be obtained by using Buckingham Pi theorem. Using this theorem, three dimensionless parameters, such as specific speed, specific capacity, and specific head can be obtained for the behaviour of a centrifugal pump.

As another empirical approach for deriving scaling parameters of local phenomena, existing correlations and models available in the literature can be used to derive scaling parameters or to estimate scaling distortions. For example, in horizontal flow, it is well known that the stratified-to-non-stratified flow regime transitions are governed by the Froude number. The derivation of the scaling parameter, i.e. the Froude number, is based on the flow regime transition correlations such as those formulated by Taitel & Dukler, 1976. From their correlations on the counter-current flow limitation, the Wallis and Kutateladze parameters usually are employed as the scaling parameters of the counter current flow limitation phenomena. The problem of this approach lies with the fact that correlations and models often do not represent the local phenomena expected in the full-scale prototype. Correlations and models are mostly developed in steady-state and/or well-developed conditions while transient and/or developing conditions may be expected during an anticipated accident scenario of the nuclear power plants. Also, the correlations and models often are developed based on experimental data that are collected using other working fluids, non-prototypical conditions, and/or using small geometries. For example, two-phase flow regime maps have been established mostly based on experimental data using air-water in low-pressure conditions for pipes of small diameter lastly, the correlations and models are mostly developed based on well isolated and well controlled experimental conditions. On the contrary, thermal-hydraulic phenomena in the full scale prototype during transient accidents may have significant interactions and influences from other local phenomena.

When the governing equations are known, the scaling groups can be derived by non-dimensional governing equations. These equations have dimensionless scaling groups. If the scaling groups with the dimensionless initial and boundary conditions are preserved between the small scale model and the prototype, the former is assumed to be similar to the large-scale prototype. However, it should be remembered that the governing equations represent an assumption of the given local phenomena, Levin et al., 1990. Uncertainties exist still in the governing equations of two-phase flow systems, two-phase flow and heat transfer correlations and the flow regime transition criteria. Thus, the scaling groups for a given local phenomena can be derived depending on the relevant assumptions and uncertainties. Experiments at different scales are necessary to validate the scaling groups of the given local phenomena.

In some cases, it is difficult to completely preserve the scaling groups between the small scale model and the prototype. For example, for horizontal flows with a free surface (separated flow, wavy flow, etc.); both the Reynolds- and Froude-numbers cannot be preserved because they require different velocity scales. In such conditions, inevitably the scaling distortions exist.

Scaling local phenomena is closely related to integral effect tests (IETs) as well as separate effect tests (SETs). SETs usually are carried out for attaining an understanding of, and for developing and validating the physical models/correlations for local phenomena. The scaling groups play an important role in the design and construction of SETFs. NEA/CSNI, 1993, relates major local phenomena to SETFs, especially for two-phase flows that may appear during LOCAs and thermal-hydraulic transients in LWRs. The scaling of local phenomena plays an important role in the bottom-up scaling for ITFs via the assessment of similarity of the local thermal-hydraulic processes and phenomena, and of scaling distortions. Scaling methods of three-level scaling and hierarchical two-tiered scaling (H2TS), for example, require the bottom-up scaling, i.e. the scaling of key processes and phenomena, as a part of the whole scaling procedure, Ishii & Kataoka, 1983, Zuber et al., 1991, and Zuber et al., 1998.

Good examples of the scaling of the local phenomena can be found in Zuber report, Zuber, 1980. It deals with two-phase flow phenomena, such as transitions in two-phase flow patterns, liquid entrainment in break flow, vapor pull-through, and counter-current flow limitations. In general, the Froude numbers in horizontal pipes, such as the hot leg and the cold leg are preserved for the similarity of the counter-current flow, transient regimes, and stratified flows. The length (L) and the diameter of horizontal pipes should be sized to preserve the volume-scaling ratio and the Froude number, important in preserving the two-phase flow regime transitions in horizontal pipes.

The full-scale Upper Plenum Test Facility (UPTF) was used to investigate the local phenomena in the primary system of pressurized water reactors during LOCA transients, Glaeser & Karwat, 1993, and Glaeser, 1992. The experimental programme includes several local phenomena, such as countercurrent flow limitations (CCFL) in the downcomer and the upper plenum, entrainment and de-entrainment in the upper plenum, and the limitation of countercurrent flow in the hot leg during reflux condensation. Comparing the full-scale results of the UPTF tests with those obtained from small-scale test facilities indicates that several phenomena, such as CCFLs, are dependent on the scale of the test facility. The results of the full-size tests of the UPTF show relatively more multi-dimensional flow behaviour, i.e. asymmetric heterogeneous, compared with the small-scale test facility. Thus, those local phenomena in the latter cannot be extrapolated to the full-sized prototype (see also the discussion in Chapter 4).

Ishii et al., 1994, derived scaling parameters for the phenomena of corium dispersion phenomena in direct containment heating (DCH). They used a step-by-step scaling approach wherein the scaling analysis was carried out by starting from the most dominant process, and considering its various mechanisms.

Yun et al., 2004, developed a modified linear-scaling law for a direct ECC bypass phenomena during the LBLOCA reflood phase. The new scaling law has the same geometrical similarity to that of the linear-scaling law, i.e. the preservation of the aspect ratio. However, the Wallis-type dimensionless parameter was chosen to be preserved for the velocity scaling of the steam and the ECC water. The velocity and time scales are reduced according to the square root of the length scale, while gravity effect is preserved. The new scaling method has been validated against experimental data obtained in test facilities of various scales including a full-scale UPTF test facility. The validation of the scaling laws showed the appropriateness of the modified linear-scaling methodology for interpreting multi-dimensional flow phenomena, such as direct ECC bypass in the reflood phase of LBLOCA. By applying modified linear scaling, the width of liquid spreading on the core barrel, the level of the onset of liquid entrainment , the direct ECC bypass fraction, and inlet sub-cooling of ECC direct bypass are well preserved in small-scale test facilities compared to full-scale UPTF test results. The power-to-volume scaling shows distortions of the circumferential gas-flow field, whilst linear scaling shows distortions in the gravity effect. Thus, the conventional scaling methods have an unavoidable excessive amount of ECC bypass in the scaled-down facilities, Song, 2006.

3.1.2 System phenomena

To assure that the transient behaviours in the scaled-down test facility are relevant to the prototype, i.e. nuclear power plants, it is necessary to develop a proper scaling-method for the thermal-hydraulic transient processes between the prototype and the scaled-down test facility. The main objectives of the scaling methods are to preserve the geometric, kinematic and dynamic similarities between thermal-hydraulic phenomena / processes that may occur in the prototype, and in the scaled-down test facility. The scaling criteria derived in the scaling method are used to design the scaled-down facility, to specify the initial- and boundary- test conditions, and finally to scale-up processes from the scaled-down test facility to the prototype.

Most scaling laws for the thermal-hydraulic phenomena are derived from the non-dimensional governing equations. For example, using three-dimensional conservation equations, Nahavandi et al., 1979, derived several set of scaling laws, i.e. time-reduction and time-preservation scaling laws. Ishii & Kataoka, 1983, derived scaling criteria using one-dimensional conservation equations for a natural circulation loop under single- and two-phase flow conditions.

For integral test facilities, after the global scaling has been settled, another level of scaling is carried out to preserve the important local thermal-hydraulic phenomena/processes, and to reduce scaling distortions. Such distortions of the local phenomena occur due to the difficulty in matching the local scaling criteria with the global ones and our lack of understanding on the local phenomenon itself, Ishii et al., 1998. Closure equations are used to generate the scaling groups. The scaling analysis focuses on various important local phenomena, closure laws, and their impacts on the system overall behaviour. The scaling typically is performed for each phenomenon in each component. The local phenomena scaling provides the estimate of possible scaling distortions and possible measures to minimize them.

The important local phenomena and processes can be identified from the phenomena identification and ranking table (PIRT). Having so identified them, scaling analysis can be performed for the major local phenomena. The closure equations for the important local phenomena are used to develop the similarity criteria to preserve the local phenomena. The bottom-up scaling analysis of the local phenomena offers the similarity parameters for scaling individual processes and phenomena of importance to the system behaviour. However, the closure equations mostly are empirical ones based on steady-state and fully developed flow conditions.

In general, it is practically impossible to preserve all the scaling criteria that govern the relevant thermal-hydraulic phenomena between the prototype and the scaled-down test facility. It is necessary to optimize the similarities for the important thermal-hydraulic phenomena and processes, and to minimize scaling distortions in the scaled-down test facility. Thus, identifying the relevant thermal-hydraulic phenomena and processes, and selecting an appropriate scaling method are essential for the scaled-down test facility. The scale-up application from the scaled-down test facility to the prototype also depends on the scaling methods used to design and operate test facility.

The scaling methods are classified into different cases depending on the choice of the major scaling parameters, i.e. pressure, power, velocity, time scale, and geometric characteristics, such as height. Each scaling method has its own distinctive advantages and also some inherent limitations that mostly are unavoidable. Deficiencies appearing inherently in each scaling methodology should be taken into account when analysing the response of a test facility and the scale-up capabilities and limitations of test results for the prototype. This naturally leads to requirements for validation of code scale-up capability.

This sub-section describes the most commonly used scaling methods to investigate the integral-effect tests in prototypes. The most important discussed methods are: linear scaling, power-to-volume scaling, three-level scaling, hierarchical two-tiered (H2TS) scaling, power-to-mass scaling, modified linear scaling, fractional scaling analysis (FSA) and dynamic system scaling (DSS). The descriptions mostly include such items as their major characteristics, merits, limitations, and areas of application.

3.1.2.1 Linear scaling

Carbiener & Cudnik, 1969, and Nahavandi et al., 1979, independently used different equations to develop a linear scaling method, and they obtained an identical similarity requirement for designing an integral effect test facility. The key characteristics of this method are to have the same aspect ratio as the prototype, and to maintain the same velocity. Similarity requirement of the main parameters in the linear scaling method under the same fluid and same operation conditions, are summarized in Table 3-1.

When a test facility is reduced by a length scale $\mathrm{l_R}$ compared to a prototype, the scaling ratio for the flow area ($\mathrm{a_R}$) is equal to $\mathrm{l_R^2}$, and the volume ratio is given as $\mathrm{V_R = l_R^3}$. For this scaling method, the scaling ratio of the acceleration rate and heat flux is inversely proportional to the length scale ($\mathrm{g_R = l_R^{-1}}$ and $\mathrm{q''_R = l_R^{-1}}$). This means that the linear scaling method can excessively increase the effect of gravitational acceleration, which results in a scaling distortion during an accident simulation when gravity effect is important.

When the effect of gravity is relatively smaller than the pressure drop in the system, such as in a blowdown phase of an LBLOCA (Large Break Loss of Coolant Accident), a water-hammer phenomenon, and an accidental steam discharge, the scaling distortion due to the increased gravity effect is negligible.

However, in a case where a static head of fluid is significant, such as a flashing by a pressure decrease, a phase separation, instability in a steam generator and reflooding in the core during the LBLOCA, a scaling distortion inevitably occurs in the linear scaling method. Also, the excessive heat flux in the core can distort the amount of the void generation and the flow-pattern transition in the reactor core.

Table 3-1 - Comparison of main scaling ratios of each scaling method.

Parameter	Symbol	Parameter Ratio (model/prototype)		
		Linear scaling	Volume scaling	Three-level scaling
Length	l_R	l_R	1	l_R
Diameter	d_R	l_R	d_R	d_R
Area	a_R	${l_R}^2$	${d_R}^2$	${d_R}^2$
Volume	V_R	${l_R}^3$	${d_R}^2$	$l_R a_R$
Core ΔT	ΔT_R	-	1	1
Velocity	u_R	1	1	${l_R}^{1/2}$
Time	t_R	l_R	1	${l_R}^{1/2}$
Gravity	g_R	$1/l_R$	1	1
Power / volume	${q_R}'''$	$1/l_R$	1	${l_R}^{-1/2}$
Heat flux	${q_R}''$	$1/l_R$	1	${l_R}^{-1/2}$
Core power	q_{Ro}	${l_R}^2$	${d_R}^2$	$a_R {l_R}^{1/2}$
Rod diameter	D_R	1	1	1
Number of rods	n_R	${l_R}^2$	${d_R}^2$	a_R
Flow rate	$\dot{m}_R$	${l_R}^2$	${d_R}^2$	$a_R {l_R}^{1/2}$
Δi subcooling	Δi_{subR}	1	1	1
ΔT subcooling	ΔT_{subR}	1	1	1

3.1.2.2 Power-to-volume scaling

The power-to-volume scaling (or volume scaling) method was suggested by Nahavandi et al., 1979. It conserves the time, length (or height), velocity, and heat flux equivalently to those of the prototype. As shown in the scaling ratios listed in Table 3-1, a reduced test facility has a full-height scale ($l_R = 1$) according to the power-to-volume scaling method, and the area and volume of the facility are reduced with the same scale ($a_R = V_R = d_R^2$). Different from the linear-scaling method, the power-to-volume scaling method can preserve the effect of gravity, so that it has an advantage in simulating those thermal-hydraulic phenomena in which the effect of gravity is significant. Therefore, it is suitable to simulate an accident in which flashing occurs by a decrease in pressure , and it has been widely used to design integral-effect test facilities such as LOFT, Ybarrondo et al., 1974, SEMISCALE, Larson et al., 1980, LOBI, Addabbo et al., 2012, PKL, Umminger et al., 2012, LSTF, ROSA-IV Group, 1985, and BETHSY, Deruaz et al., 1982. Also, it can be successfully applied to the heat transfer test in an electric heater bundle simulating nuclear fuel, and a critical heat-flux test.

However, when the power-to-volume scaling method is applied to a test facility with a too small area scale, major phenomena can be distorted significantly. Especially, pressure drops and heat losses of the system, and accumulated heat of test facility structures, become excessive in the small scale facility. Also, the aspect ratio (l_R/d_R) is increased due to the reduced area at the full-height condition, and it makes inadequate simulation in the test facility for the multidimensional flow phenomenon

3.1.2.3 Power-to-mass scaling

Scaling parameters from the linear-scaling method, the volume-scaling method, and the three-level scaling method are derived, assuming that the test facility can simulate the equivalent pressure and temperature conditions as the prototype. It means that those scaling methods are available in a full-pressure test facility, which can preserve the properties of the working fluid.

To determine the test conditions of a reduced-height, reduced-pressure (RHRP) facility, the power-to-mass scaling method was developed and applied to perform an integral effect test in the IIST (INER Integral System Test) facility in Taiwan, Liu et al., 1997, and Liu et al., 1998. The facility is a scaled model with approximately 1/400 in a volume to simulate the Maanshan Pressurized Water Reactor (MPWR). The normal operation pressure of the facility is reduced to 2.1 MPa. Since the fluid properties and the mass inventory cannot be equivalently preserved under conditions of reduced pressure, the scaling parameters for the core power (${l_R}^2$ in the linear scaling method, ${d_R}^2$ in the volume scaling method, $a_R {l_R}^{1/2}$ in the three-level scaling method) are not directly applicable to the RHRP integral effect test facility.

The power-to-mass scaling methodology determines the thermal power in the core, (Q), according to the initial inventory of coolant mass in the reactor coolant system (M_o), as follows:

$$\left[\frac{Q-Q_h}{M_o}\right]_m = \left[\frac{Q}{M_o}\right]_p, \tag{3-1}$$

Where Q_h represents the heat loss of the test facility. The hot leg temperature (T_{HL}) in the test facility is determined from the degree of subcooling relative to the saturation temperature of the primary system, ($T_{sat}(p_0)$):

$$(T_{sat}(p_0) - T_{HL})_m = (T_{sat}(p_0) - T_{HL})_p \tag{3-2}$$

From the hot-leg temperature, the temperature of the cold leg is determined to equivalently scale the difference in core temperature, as shown in Eq. (3-3):

$$(T_{HL} - T_{CL})_m = (T_{HL} - T_{CL})_p \tag{3-3}$$

To achieve the criteria in Eq. (3-3), the mass flow rate of the core ($\dot{m}$) should be scaled down according to the following relationship:

$$\frac{\dot{m}_m}{\dot{m}_p} = \frac{[(Q-Q_h)/C_p]_m}{(Q/C_p)_p} \tag{3-4}$$

Finally, secondary system pressure (p_s) is determined according to the following relationship;

$$(T_{CL} - T_{sat}(p_s))_m = (T_{CL} - T_{sat}(p_s))_p \tag{3-5}$$

To validate applying the power-to-mass scaling method to the IIST facility, counterpart tests were conducted for an SBO, and a cold leg SBLOCA. The test results were compared to the experimental data from the ROSA-IV LSTF (1/48 scale by volume) and BETHSY (1/100 scale by volume), both of which are full-height full-pressure (FHFP) test facilities, Liu et al., 1997, and Liu & Lee, 2004. In the counterpart test of the IIST for the SBO transient, the major thermal-hydraulic responses, such as the secondary coolant boil-off, and the subsequent primary-coolant saturation, pressurization, the depletion and redistribution of the inventory of coolant, and the uncovering of the core caused by boil-off of the coolant were found in good agreement with the result of the LSTF. Counterpart tests of the IIST and BETHSY for the SBLOCA scenario also proved that the power-to-mass scaling method could successfully preserve the major phenomena, including the loop seal clearing and associated decrease of the core level in the RHRP test facility.

3.1.2.4 Hierarchical Two-Tiered Scaling (H2TS)

The Hierarchical, two-tiered scaling method, the H2TS method, was developed to provide a comprehensive and traceable scaling-methodology and to minimize arbitrariness in deriving the scaling requirements, Zuber et al., 1991, (App. D in NUREG/CR-5809), and Zuber et al., 1998. H2TS was influenced by the theory of hierarchy, Simon, 1962. This paper is the foundation for applying the hierarchy theory in ecology and in complex-systems theory in general. Kenneth Bounding's "Bathtub Theorem" is often referred to in modelling differentiated aggregates, Weinberg & Weinberg, 1988.

H2TS was successfully adopted to design the APEX test facility, Reyes et al., 1998. It also has been employed in EMDAP (Evaluation Model Development and Assessment Process) and CSAU (Code, Scaling, Applicability and Uncertainty), methods that evaluate uncertainty in the safety analysis, USNRC, 2005. Procedures for H2TS scaling methods are composed of four stages, i.e. system breakdown, scale identification, top-down and bottom-up scaling analysis.

In the first stage, the system is decomposed into subsystems, modules, constituents, phases, geometrical configurations, fields, and processes. This architecture of the decomposed system is used to establish hierarchies for three measures that characterize important transfer processes, i.e. the volumetric concentration (α), the spatial scale (L), and the temporal scale (τ). α Is the volume fraction of a given constituent or phase, L is related to the scale of the transfer area for a given process, and τ is the governing parameter for the rate of transfer.

The second stage of the H2TS scaling method provides a hierarchy for characteristic volume fraction, spatial scale, and temporal scale. The volumes of the control volume (V_{CV}), constituent (V_C), phase (V_{CP}), and geometrical configuration (V_{CPG}) are related by the volume fractions, α_C, α_{CP}, and α_{CPG}. In the case of the hierarchy for characteristic spatial scales, the ratio of the transfer area (A_{CPG}) for a specific process to the volume (V_{CPG}) is defined with a characteristic length scale (L_{CPG}) as follows;

$$\frac{A_{CPG}}{V_{CPG}} = \frac{1}{L_{CPG}} \qquad (3\text{-}6)$$

To establish the hierarchy of the temporal scale, the characteristic frequency of a specific process across an area $A_{CPG}(\omega_{CPG})$ is defined in Eq. (3-7), wherein ψ is a property (mass, momentum, energy) per a unit volume, and j_i is the flux of ψ. ω_{CPG} can be related to the characteristic frequency in the control volume $V_{CV}(\omega_i)$ as shown in Eq. (3-8). From the characteristic frequency of each process, the characteristic time ratio (Π_i) is defined in Eq. (3-9), using the system response time ($\tau_{CV} = V_{CV}/Q_f$, where Q_f is a volumetric flow rate). These are shown here:

$$\omega_{CPG} = \frac{j_i A_{CPG}}{\psi V_{CPG}} \qquad (3\text{-}7)$$

$$\omega_i = \frac{j_i A_{CPG}}{\psi V_{CV}} = \alpha_C \alpha_{CP} \alpha_{CPG} \omega_{CPG} \qquad (3\text{-}8)$$

$$\Pi_i = \omega_i \tau_{CV} = \alpha_C \alpha_{CP} \alpha_{CPG} \omega_{CPG} \tau_{CPG} \qquad (3\text{-}9)$$

The third- and fourth-stages of the H2TS scaling method are the top-down and the bottom-up scaling approach, respectively. The top-down scaling analysis is a method for establishing a scaling hierarchy, using the conservation equations of the mass, momentum, and energy in a control volume. A non-dimensional form of the balance equation for a constituent "i" can be written as follows:

$$\tau_i \frac{d(V_i^+ \psi_i^+)}{dt} = \Delta[Q_i^+ \psi_i^+] \pm \sum_{k=1}^{m-1} \Pi_{ik} j_{ik}^+ A_{ik}^+ \qquad (3\text{-}10)$$

In Eq. (3-10), many characteristic time ratios (Π_{ik}) exist for the processes between the constituent "i" and other m-1 constituents. Consequently, all processes for each constituent, phase, and geometrical configuration can be evaluated in terms of the time, and it is possible to rank them according to their importance on the system. Such a scaling hierarchy can identify similarity groups between a prototype and a scaled-down facility (model), and establish priorities for the design of the test facility, code development, and uncertainty quantification. Also, the characteristic time-ratio can be utilized to determine scaling distortion for a specific transfer-process in the test facility (model) as defined in Eq. (3-11):

$$D = \frac{[\Pi_i]_{\text{prototype}} - [\Pi_i]_{\text{model}}}{[\Pi_i]_{\text{prototype}}} \tag{3-11}$$

The bottom-up scaling approach is a detailed scaling analysis for key processes and phenomena. In this stage, the important phenomena in a subsystem are identified, and the sequence of analysis for the processes and the mechanisms are determined. Then, applying a step-by-step integral method for the processes, the scaling criteria and time constants are obtained. Finally, the relative importance of the processes can be evaluated.

3.1.2.5 Three-Level Scaling

Ishii & Kataoka, 1983 suggested the three-level scaling method that focuses on the conservation of natural circulation commonly occurring in most design-basis accidents. Since this scaling method is beneficial in designing a test facility with different ratios of height and area, it is suitable for a design of a test facility with reduced height. Hence, this scaling method has contributed in designing and constructing the integral-effect test facilities, such as PUMA, Ishii et al., 1998, and ATLAS, Kim et al., 2008, and Choi et al., 2014. It consists of the following three scaling analysis steps, Ishii & Kataoka, 1998.

The first step is an integral analysis or a global-scaling analysis to conserve the single- and two-phase natural circulation flow. The similarity requirement is obtained from non-dimensional form of the governing equations of natural-circulation flow. Global similarity parameters for single- and two-phase natural circulation were derived from the equations for fluid continuity, integral momentum, and energy in one-dimensional, area-averaged forms, along with their appropriate boundary conditions, and the solid structure energy equation. At this step, general similarity parameters related to the macroscopic behaviour of the whole system are conserved in the test facility, and the geometric requirement, time scale, and similarity requirement of the main thermal-hydraulic parameters are determined. Table 3-1 summarizes the scaling parameters of the three-level scaling methods under the same fluid- and operational-conditions (pressure and temperature).

For a single-phase flow, one-dimensional area average continuity, integral momentum and energy-equations are used. First, the relevant scales for the basic parameters are determined, and then, the similarity groups are obtained from the conservation equations and boundary conditions. The heat transfer between the fluid and structure are included in the analysis, using the energy equation for the structure. From these, important dimensionless groups are derived as follows:

Richardson number, $$R \equiv \frac{g\beta\Delta T_o l_o}{u_o^2} = \frac{\text{Buoyancy}}{\text{Inertia force}} \tag{3-12}$$

Friction number, $$F_i \equiv \left[\frac{f_w l}{d} + K\right]_i = \frac{\text{Friction}}{\text{Inertia force}} \tag{3-13}$$

Modified Stanton number, $$St_i \equiv \left[\frac{4hl_o}{\rho_f c_{pf} u_o d}\right]_i = \frac{\text{Wall convection}}{\text{Axial convection}} \tag{3-14}$$

Time ratio number, $$T_i^* \equiv \left[\frac{l_o / u_o}{\delta^2 / \alpha_s} \right]_i = \frac{\text{Transport time}}{\text{Conduction time}} \quad (3\text{-}15)$$

Biot number, $$B_{ii} \equiv \left[\frac{h\delta}{k_s} \right]_i = \frac{\text{Wall convection}}{\text{Conduction}} \quad (3\text{-}16)$$

Heat source number, $$Q_{si} \equiv \left[\frac{q_s^{'''} l_o}{\rho_s c_{ps} u_o \Delta T_o} \right]_i = \frac{\text{Heat source}}{\text{Axial energy change}} \quad (3\text{-}17)$$

Pump characteristic number, $$F_d \equiv \frac{g \Delta H_d}{u_o^2} = \frac{\text{Pump head}}{\text{Inertia}} \quad (3\text{-}18)$$

Axial length scale, $$L_i \equiv \frac{l_i}{l_o} \quad (3\text{-}19)$$

Flow-area scale, $$A_i \equiv \frac{a_i}{a_o} \quad (3\text{-}20)$$

Where the subscripts i, f, and s, respectively, mean the i-th component of the loop, fluid, and solid. Here, u_o, ΔT_o, l_o, and a_o are the reference velocity, temperature difference across the core, equivalent length (heated length) and equivalent flow area. The conduction depth is defined by:

$$\delta_i \equiv a_{si} / \zeta_i \quad (3\text{-}21)$$

Where a_{si} and ζ_i are the solid structure cross-sectional area and wetted perimeter of the i-th section. The pump characteristic number was added in the dimensionless groups so to consider the forced circulation flow.

For a two-phase natural circulation system, similarity groups were developed from a perturbation analysis based on the one-dimensional drift flux model, Ishii & Kataoka, 1983. The four-equation drift-flux model consisting of mixture mass, vapor mass, momentum, and energy equations are integrated along the flow path. The integral transfer functions between the inlet perturbation and various variables are obtained. These transfer functions are cast in non-dimensional forms to yield the following two-phase similarity parameters:

Phase-change number (Zuber number), $$N_{pch} \equiv \left[\frac{4 q_o^{'''} \delta l_o}{d u_o \rho_f i_{fg}} \right] \left[\frac{\Delta \rho}{\rho_g} \right] \quad (3\text{-}22)$$

Sub-cooling number, $$N_{sub} \equiv \left[\frac{\Delta i_{sub}}{i_{fg}} \right] \left[\frac{\Delta \rho}{\rho_g} \right] \quad (3\text{-}23)$$

Froude number, $N_{FR} \equiv \left[\frac{u_o^2}{g l_o \alpha_o}\right]\left[\frac{\rho_f}{\Delta\rho}\right]$ (3-24)

Drift-flux number (or void-quality relation), $N_{di} \equiv \left[\frac{u_{gj}}{u_o}\right]_i$ (3-25)

Time-ratio number, $T_i^* \equiv \left[\frac{l_o / u_o}{\delta^2 / \alpha_s}\right]_i$ (3-26)

Thermal-inertia ratio, $N_{thi} \equiv \left[\frac{\rho_s c_{ps} \delta}{\rho_f c_{pf} d}\right]_i$ (3-27)

Friction number, $N_{thi} \equiv \left[\frac{f_w l}{d}\right]_i \left[\frac{1 + x(\Delta\rho / \rho_g)}{(1 + x\Delta\mu / \mu_g)^{0.25}}\right]\left[\frac{a_o}{a_i}\right]^2$ (3-28)

Orifice number, $N_{oi} \equiv K_i \left[1 + x^{3/2}(\Delta\rho / \rho_g)\right]\left[\frac{a_o}{a_i}\right]^2$ (3-29)

Similarity can be achieved between the processes expected to occur in the prototype and those in a model provided that the above dimensionless groups of the model are the same in the prototype. Table 3.1 summarizes the scaling parameters of the three-level scaling method under the same fluid and operational conditions (pressure and temperature).

The second step is scaling of boundary flow and inventory. For a system consisting of several inter-connected components, a proper scaling of the inter-component relations is important in preserving the thermal-hydraulic interactions between these components. The scaled mass and energy inventories for each component can be obtained from the control-volume balance equations for mass and energy. At the interface between two connected components, the scaling criteria are obtained from the boundary mass and energy flows. The discharge-flow phenomena at the breaks and at the safety- and depressurization-valves should be preserved to assure similar depressurization histories between the prototype and the model. To maintain similar overall behaviour between the prototype and the model, the depressurization histories should be the same when compared in their respectively scaled time-frames. A separate scaling criterion for the system-boundary flows, such as the break flow and various ECCS injection flows, can be obtained from the dimensionless mass-conservation equation.

In the last step, a local phenomenon scaling is performed to conserve the important thermal-hydraulic phenomena occurring in each system. Even though the overall similarity of the system response is achieved from the integral scaling step, the needed local thermal-hydraulic phenomena in a specific component can remain unsatisfied. In this step, a local similarity analysis on the key thermal-hydraulic phenomena in the system is covered. If the similarity requirement derived at this step (local phenomena scaling analysis) is different from that in the integral scaling, the requirement for the latter is replaced by the result from scaling of local phenomenon to conserve the physical phenomena with higher priority.

The three-level scaling method is characterized by relaxed restriction in the length scale. By adopting a proper length scale, the scaling distortion of a comparatively small-scale test facility can be minimized when compared to the power-to-volume scaling method. Since the aspect ratio of a test facility with the three-level scaling method is closer to the prototype, multidimensional phenomena can be conserved reasonably even in a relatively small-scale facility. On the other hand, the scales for time and flow velocity are reduced due to the reduced length-scale ($t_R = l_R^{1/2}$ and $u_R = l_R^{1/2}$), so that local thermal-hydraulic phenomena inevitably are distorted. This distortion happens similarly for the previous H2TS. The distortion can be overcome by satisfying the similarity requirement from the local-phenomena scaling at the third step.

3.1.2.6 Modified linear scaling

As a Direct Vessel Injection (DVI) system was adopted instead of the conventional Cold Leg Injection (CLI) system in APR1400 (Advanced Power Reactor 1400 MWe), multi-dimensional behaviour of the ECC (Emergency Core Cooling) water in a downcomer, such as a direct ECC bypass or a sweep-out, can be observed during the reflood phase of the LBLOCA.

A modified linear scaling method was suggested to investigate the direct ECC-bypass phenomena in a small-scale test facility, Yun et al., 2004. Similarity parameters were derived from the two-dimensional continuity and momentum equations of a two-fluid model. This yielded twelve dimensionless groups, which were obtained from the two-fluid momentum equations for the liquid and the gas phases in the downcomer. Table 3-2 shows the twelve similarity groups. To preserve those groups between the prototype and the test facility, the similarity requirements were derived as shown in Table 3-3, and compared to the linear scaling method. This shows that the modified linear scaling method requires the same geometry similarity criterion with the linear scaling method; however, the modified method can preserve gravity effect.

Table 3-2 – Similarity parameters of the modified linear-scaling method.

Similarity Parameters	
$\pi_1 = t_o j_{xko}/\alpha_{ko} L_o$	$\pi_7 = j_{yko}/j_{xko}$
$\pi_2 = t_o j_{yko}/\alpha_{ko} L_o$	$\pi_8 = 1/S_{xo}$
$\pi_3 = \alpha_{ko} t_o g_o/j_{xko}$	$\pi_9 = 1/S_{yo}$
$\pi_4 = \alpha_{ko} t_o \Delta p_{xo}/j_{xko}\rho_{ko} L_o$	$\pi_{10} = \alpha_{ko} t_o \Delta p_{yo}/j_{yko}\rho_{ko} L_o$
$\pi_5 = f_{wxko} j_{xko} t_o/L_o \alpha_{ko}$	$\pi_{11} = f_{wyko} j_{yko} t_o/L_o \alpha_{ko}$
$\pi_6 = f_{ixo}\rho_{go} j_{xgo}^2 t_o/L_o \alpha_{ko}^2 \rho_{ko} j_{xko}$	$\pi_{12} = f_{iyo}\rho_{go} j_{ygo}^2 t_o/L_o \alpha_{ko}^2 \rho_{ko} j_{yko}$

Table 3-3 – Scaling parameters of the modified linear scaling method, Yun et al., 2004.

Parameter	Symbol	Parameter Ratio (model/prototype)	
		Linear scaling	Modified linear scaling
Length	l_R	l_R	l_R
Area	a_R	${l_R}^2$	${l_R}^2$
Volume	V_R	${l_R}^3$	${l_R}^3$
Velocity	u_R	1	${l_R}^{1/2}$
Time	t_R	l_R	${l_R}^{1/2}$
Gravity	g_R	$1/l_R$	1
Flow rate	$\dot{m}_R$	${l_R}^2$	${l_R}^{5/2}$
Temperature	T_R	1	1
Void ratio	α_R	1	1
Slip ratio	S_R	1	1
Aspect ratio	l_R/d_R	1	1

From a comparison with the similarity requirements in Table 3-1, it is found that the three-level scaling method can provide the same requirements with the modified linear scaling method when the aspect ratio is preserved in a test facility: $a_R = {l_R}^2$. The three-level scaling method focused on preserving the natural circulation-flow, while the modified linear scaling-method was derived to preserve the multi-dimensional thermal-hydraulic phenomena in the reactor vessel downcomer during the re-flooding phase of the LBLOCA. By comparing the length of the liquid film and the direct bypass fraction from the experimental data from facilities with different scales, Yun et al., 2004, found that the modified linear scaling method successfully preserved the multi-dimensional ECC-bypass phenomena.

3.1.2.7 Fractional Scaling Analysis (FSA)

Fractional Scaling Analysis (FSA), Zuber et al., 2007, Catton et al., 2009, and Wulff et al., 2009, was developed as advancement from H2TS as the earlier method, Zuber et al., 1990. FSA is based on well-established general theory (see, for example Novozhilov, 1997).

FSA is a systematic method of ranking components and the phenomena in the components in terms of their effect on the figure of merit (FOM) or safety parameter, of estimating scale distortions, and also a way to synthesize data from different facilities for the same class of transients. This multistage scaling will also guide in designing a scaled facility by identifying important components and their corresponding important processes. The scaling process will help in simplifying the facility design by providing flexibility in addressing only the important components.

As pointed out by Zuber, 2005, the FSA approach was developed for and applied to complex situations. It is very general method, and has been applied to optimizing ecology-related decisions, e.g. Allen & Starr, 1982. In such situations, the problem is to understand the system features based on the knowledge of the effects and/or the system performance. As an illustration, FSA was also applied to system level for LOCA analyses, Wulff et al., 2009, and at component level for thermal analyses of fuel rods, Catton et al., 2009, (see also Catton et al., 2005). FSA can assist with nuclear thermal-hydraulics for which predictive capabilities are available, and the system fundamental features are known (see also discussion at the end of this Section).

As the first step in an FSA, the region of interest and duration of transients are identified. The region or system is made of connected components, and their thermal-hydraulic performance is result of processes. The state variables over the region are connected to the transfer functions at the boundary and inside the volume. The relative effect of components is based on their relative impact on state variable from the transfer function connected to that component. The relative effect is determined from making the transfer terms non-dimensional. The relative values determine the importance of the transfer terms. The important terms can be investigated further by looking at the contributions from important components to identify the scaling groups and the important processes.

The performance of facility is governed by the balance equations, boundary conditions, and initial conditions. The state variables, for a given control volume, change in response to processes or agents of change (AOCs) that are taking place, inside and at the boundaries. The balance equation (integral form) is made non-dimensional with reference or characteristic values of the state variables and parameters forming the AOC, such that the time-dependent components are the order of one, and the values of the coefficients represent the characteristic magnitude of the AOC.

In well-instrumented facilities wherein the AOC can be measured over time, the relative impact on the state variable will be an accurate representation of the Pi groups, and the total impact of each AOC can be estimated. Lacking such information, the most reliable known quantities over the transient are used to estimate the Pi groups (also for designing). Reliable and applicable codes also can provide the information needed to estimate relative impact of AOCs on the state variables.

The fractional change of the state variable (control volume) over a characteristic time should be made the same for two facilities (fractional change metric) for top-level scaling. Characteristic time is obtained either from the experiments or from aggregate fractional rate-of-change (also called aggregate frequency). Each agent of change contributes to fractional change through fractional rate of change (FRC), ωI, and characteristic time, tref. This change could be positive or negative. The reference value of magnitude of the AOC should be maximum value over the period of phase. This could be initial value at the beginning of the phase or some set-point during the phase.

The analytical derivation of the FSA approach can be found in Zuber et al., 2007, Catton et al., 2009, and Wulff et al., 2009. Hereafter, key definitions are reported to clarify the summary description of the method.

In the case of a region of space characterized by a state variable F (or a single module), undergoing a change caused by a single AOC, identified by φ, one may write (for instance, if F is the energy, φ is the power):

$$\frac{dF}{dt} = \varphi \tag{3-30}$$

Then, the fractional rate of change (FRC), ω, the characteristic length, λ, and the fractional change, Ω, also called effect metrics, are defined, respectively as follows:

$$\omega = \frac{1}{F}\frac{dF}{dt} \tag{3-31}$$

$$\frac{1}{\lambda} = \frac{A}{F} \tag{3-32}$$

$$\Omega = \omega\, t\,, \tag{3-33}$$

Where t is a characteristic time for the process, and can be either the physical transient time (‘clock time’, following Zuber, 2005), or a time derived by using ω (i.e. a process-specific time constant), and A is the signal transfer area, or the flow area in the case of a pipeline. It is interesting to note the following:

- The characteristic length λ in the case of blow-down analysis during LOCA is defined by $\lambda = V/A_{break}$, where $V \equiv F$ and A_{break} is the break area, assuming no condition of critical flow occurs inside *V*. When the concept of influence volume and heat transfer area is not identifiable, the characteristic length can be defined as the length required completing the process of change.
- Each change in the state variable, *F*, is associated with convection, or diffusion, or wave propagation.
- In nuclear thermal-hydraulics (i.e. when the balance equations are considered) Ω will correspond to any key non-dimensional quantity like, Re, Fr, and Bi.
- The ‘paradigm of FSA’ is that processes having the same Ω are expected to be similar because their state variables have been changed by the same fractional amount. This implies that similarity requires only the equality of Ω, and FRC (or ω) and the clock time need not be preserved.

A single module characterized by the state variable, F, and acted upon a single AOC φ (hence the corresponding FRC, ω) can be considered as a first-level element in building a complex aggregate of interacting modules, Zuber, 2005. Then, in the case of aggregates consisting of ‘j’ interconnected or interacting modules (these may correspond to control volumes in a lumped-parameter model approach in nuclear thermal-hydraulics), acted upon by ‘i’ AOC, the formulation of FSA becomes;

$$\frac{dF}{dt} = \sum_i \phi_i \tag{3-34}$$

$$\frac{dF^*}{dt^*} = \sum_i \phi_i^* \phi_{i,ref} t_{ref} / F_{ref} = \sum_i \phi_i^* \omega_i t_{ref} \tag{3-35}$$

$$\omega_i = \frac{\phi_{i,ref}}{F_{ref}} \tag{3-36}$$

$$\omega_{agg} = \sum_{i,algebr} \omega_i \tag{3-37}$$

$$\pi_i = \frac{\omega_i}{\omega_{agg}} \tag{3-38}$$

$$\Omega_i = \omega_i * t_{ref} \tag{3-39}$$

$$t_{ref} = {}^{1}/_{\omega_{agg}} \tag{3-40}$$

In the above equations, other than the already introduced symbols, ‘*’, ‘ref’ and ‘agg’ indicate a dimensionless (or a normalized value) quantity, a reference value for scaling, and a property of the aggregate system, respectively; □ gives the relative importance of an individual FRC when multiple FRC are calculated. The second equation above can be taken as the paradigm of FSA when an aggregate system is involved.

In the case of facilities with the same characteristic times, these π groups can be compared for assessing similarity and (as already mentioned) preservation of time is not essential for similarity. In the case of different characteristic times, t1 and t2, the distortion is assessed by comparing these individual fractional changes (Ωj) for two facilities:

$$\left(\Omega_j\right)_1 = \left(\omega_j\right)_1 t_1 \Leftrightarrow \left(\Omega_j\right)_2 = \left(\omega_j\right)_2 t_2 \tag{3-41}$$

The similarity cannot be guaranteed for all components or phenomena, but should be met for its important components or phenomena. This is done by arranging FRC, ω_i, or the fractional change of effect metrics, Ώi, in order of their magnitudes. The FRC is the intensity of the effect of the agent of change. As shown in Table 3-4, the first agent of change is the most important one.

The Hierarchy of Agents of Change establishes a quantitative, objective order of phenomena importance, and supersedes the subjective Phenomena Identification and Ranking Table (PIRT). This prioritizes the processes that require attention in reactor design, experiments (scaling criteria to be met), code development, and resource allocation.

Table 3-4 – Establishing the hierarchy for the agent of change.

Agent 1		Agent 2		Agent 3		Agent j, . . .		Agent n
$\lvert\omega_1\rvert$	>	$\lvert\omega_2\rvert$	>	$\lvert\omega_3\rvert$	>	$\lvert\omega_j\rvert$	>	$\lvert\omega_n\rvert$

Agent 1		Agent 2		Agent 3		Agent j, . . .		Agent n
$\lvert\Omega_1\rvert$	>	$\lvert\Omega_2\rvert$	>	$\lvert\Omega_3\rvert$	>	$\lvert\Omega_j\rvert$	>	$\lvert\Omega_n\rvert$

The characteristic time can also be estimated either by dominant FRC, ωI, or by an aggregate of FRC, ωagg. It can also be established by time it takes for two facilities to have the same fractional change in control volume parameter. It should be noted that characteristic time does not have to be preserved. As shown in Fig. 3-1, two facilities have different rates of depressurization and therefore, have different characteristic times. However, if time is made non-dimensional with characteristics times, these two curves will overlap.

Scale distortion for comparing facilities on the basis of FRCs (ω) is possible only if time is preserved between them, such as LOFT & Semiscale for LBLOCA. The comparison must be based on the Effect Metric, Ω, for Facilities 1 and 2, and for each Agent of Change, j. The quantitative assessment of scale distortion is directly estimated from the difference in fractional change of a state variable, or a safety parameter during a selected time for given component, b.

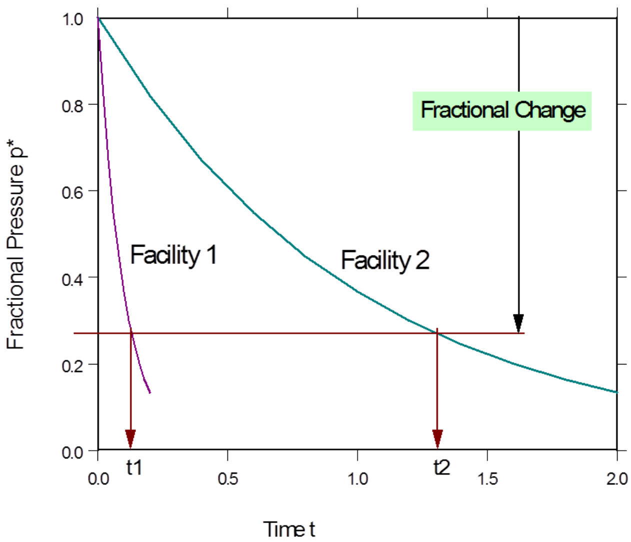


Fig. 3-1 – Application of FSA to the depressurization of two facilities.

$$\Delta\Omega_b = (\Omega_\mathrm{b})_1 - (\Omega_b)_2 \tag{3-42}$$

This can be further devolved at phenomena level for component b.

$$\Delta\Omega_{b,p1} = \left(\Omega_{\mathrm{b,ph1}}\right)_1 - \left(\Omega_{b,ph1}\right)_2 \tag{3-43}$$

Relative Scaling Analyses (RSA), a variation of FSA, was proposed for CANDU application, Wan, 2007. The difference in this approach was to use system code calculation for the plant and the facility. This provided the reference quantities such as the highest value, time period, value of state variable at the beginning of the phase and average value of agent of change, $\boldsymbol{\phi}$, over the period of the phase for scaling. A determination of the code applicability to both facilities will be needed before applying this approach. The equations used for RSA are:

$$F(t')\,\frac{dF^*(t)}{dt} = \sum_i^N \phi_i^{extremum}\;\phi_i^*(t) \tag{3-44}$$

$$\omega_i = \frac{\phi_i^{exter}}{F(t_1)} \tag{3-45}$$

$$\Delta F^* = \sum_i^N \Delta F_i^* = \sum_i^N (\omega_i\;\overline{\phi}_i^*)\;\Delta t = \overline{\omega}_{agg}\;\Delta t \tag{3-46}$$

$$\overline{\omega}_{agg} = \sum_i^N \omega_i\;\overline{\phi}_i^* = \frac{\Delta F^*}{\Delta t} \tag{3-47}$$

$$\pi_i = \frac{\Delta F_i^*}{\Delta F^*} = \frac{\omega_i\;\overline{\phi}_i^*}{\overline{\omega}_{agg}} = \frac{\omega_i\;\overline{\phi}_i^*}{\sum_i^N \omega_i\;\overline{\phi}_i^*} \tag{3-48}$$

The difference between FSA and RSA expressions is the use of dimensionless average values of agents of change, $\overline{\Phi^*}$. If these are assumed to be order of 1 or reference values used are average values, the FSA and RSA have same expressions. The Pi groups are independent of reference values of ϕ in RSA as shown by the following equation:

$$\pi_i = \frac{\omega_i\overline{\phi_i^*}}{\sum_i \omega_i\overline{\phi_i^*}} = \frac{\overline{\phi_i}}{\sum_i \overline{\phi_i}} \tag{3-49}$$

Here, we note that FSA can be defined as a logical-framework-analytical approach, suitable for applications in complex technologies (economics, ecology, Zuber, 2005). In addition, a) FSA focuses (so far) on processes wherein the state variables are influenced only by convection, diffusion and wave propagation, Zuber 2005; b) FSA has been proposed for understanding the system features based on the knowledge of effects and/or of the system performance.

One may note that the FSA application has less interest in a situation where fundamental system features are known, all dominant phenomena are identified and good predictive capabilities of nuclear thermal-hydraulics are available.

3.1.2.8 Dynamical System Scaling (DSS)

To evaluate time dependent effects of scaling there have been several attempts, e.g. Dzodzo, 2009, and Achilli et al., 2012. In these studies the non-dimensional similarity parameters (the ∏ groups) were evaluated at different time points of the transient, such as the initial condition or the average condition of a period. The main purpose is to ensure the worst scaling distortion is covered in a transient in order to justify a scaling design.

An innovative approach with similar origin of H2TS and FSA was developed recently, trying to incorporate the dynamic response of a thermal-hydraulic process into the scaling framework. This approach exploits the concept of the response of a classical dynamical system in the processes time geometry; therefore it is named Dynamical System Scaling (DSS), Reyes, 2015, and Reyes et al., 2015.

The main idea is to convert a physical process through a coordinate transformation into the phase space (coordinate) that is traditionally used in a dynamical system. The state of the physical system becomes a point (an object) in the phase space, and the entire physical process can be depicted as the trajectory of the object. By the invariance of inertial coordinate transformation in special relativity theory, the geometric similarity of two trajectories in an inertial phase-space can be viewed as a similarity of two physical processes in two clock-time systems, and vice versa. A benefit from the possibility of quantitatively evaluating in the phase space is that the phase curve can be used to describe the time-dependent effects of the physical process. In a thermal-hydraulic experiment, if a similarity is established between a model and a prototype, the phase curves of these two systems will overlap in the phase space in the entire period of the transient. Any deviation of the phase curves geometrically can be used to assess the scaling distortion of the physical processes.

The following equation is a standard balance equation for a transport process. In a confined region, V, the change of the controlled property (mass, momentum or energy denoted by Ψ(x,t)) is balanced by the property flowing across the boundary, exerted from external fields and other volumetric property sources.

$$\frac{d}{dt}\iiint_V \psi\left(\vec{x},t\right)dV = \iiint_V \left(\phi_v + \phi_f\right)dV + \iint_A \left(\vec{j}\cdot\vec{n}\right)dA - \iint_A \psi\left(\vec{v}-\vec{v}_s\right)\cdot\vec{n}\,dA = \sum_{i=1}^{n}\varphi_i \qquad (3\text{-}50)$$

By dividing a reference quantity (Ψ0), and defining new variables of β and ω,

$$\beta(t) = \frac{1}{\Psi_o}\iiint_V \psi\left(\vec{x},t\right)dV \qquad (3\text{-}51)$$

$$\omega(t) = \frac{1}{\Psi_o}\left[\iiint_V \left(\phi_v + \phi_f\right)dV + \iint_A \left(\vec{j}\cdot\vec{n}\right)dA - \iint_A \psi\left(\vec{v}-\vec{v}_s\right)\cdot\vec{n}\,dA\right] = \frac{1}{\Psi_o}\sum_{i=1}^{n}\varphi_i \qquad (3\text{-}52)$$

$$\frac{d\beta}{dt} = \omega \qquad (3\text{-}53)$$

In Eq. (3-53), the rate of change of the property is equal to all the changes brought by all transfer-processes occurring in the control region. In the meantime, the characteristic time of the entire process can be defined as a new variable, $\tau = \beta/\omega$. Through some derivations, the differential of this characteristic time, τ , with respect to real time (clock time), t can be related to differential change of β and ω

$$\frac{d\tau}{dt} = \frac{1}{\omega}\frac{d\beta}{dt} - \frac{\beta}{\omega^2}\frac{d\omega}{dt} \tag{3-54}$$

To examine the relative change of process time to real time, another variable, D, is defined as

$$D = \frac{d\tau - dt}{dt}, \tag{3-55}$$

This is related to β and ω by:

$$D = -\frac{\beta}{\omega^2}\frac{d\omega}{dt} \tag{3-56}$$

so that,

$$d\tau = (1 + D)dt \tag{3-56a}$$

D also can be used to calculate the duration of the process time by integrating it over the real time.

$$\tau_2 - \tau_1 = \int_{t_1}^{t_2} (1 + D)dt \tag{3-57}$$

After these derivations, the physical process can be plotted on a different coordinate of β and ω, the phase space, shown in Fig. 3-2. As stated before, any geometric point on the plan represents a state point of the system. There are state points with same process time, τ, that compose a line passing through the origin. On this line, $d\tau = 0$ since τ is constant, and the line is called a null geodesic. As a state point moves from $\tau 1$ to a different state point of $\tau 2$, the trajectory represents the transition of the physical process. The arc length (called the action) of the process trajectory can be evaluated as follows:

$$\tau_S = \int_{t_1}^{t_2} (1 + D)\, dt \tag{3-58}$$

It is noted that the trajectory needs not to be on a flat surface (i.e. two-dimensional one).
The beauty of this transformation lies in the principle of covariance, i.e. the physical laws remain the same if viewed in a different inertial coordinate. If two phase curves remain the same after an inertial coordinate transform in the phase space of β and ω, then the physical processes represented by these two phase curves are similar in the physical systems. In terms of scaling, this means that the thermal dynamic responses of these two systems (the prototype and the model) are the same, and the similarity is established. This provides a way to scale the process from a reactor to a test facility. According to the principle of covariance, the following relationship stands, where M denotes model, and P denotes the prototype.

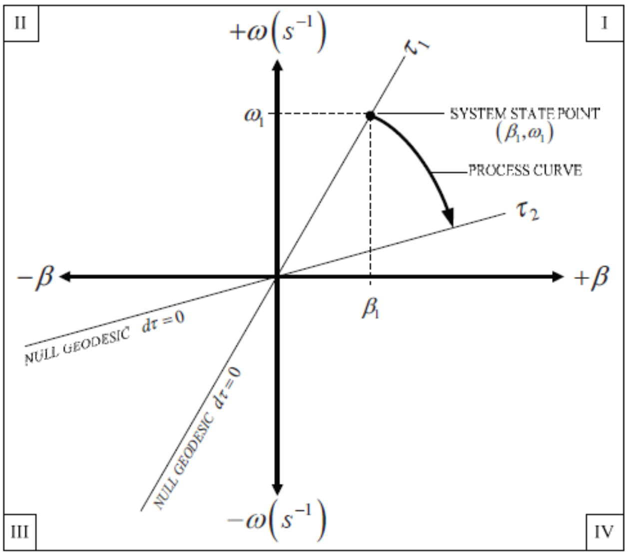


Fig. 3-2 – Working principle for DSS.

$$\frac{1}{\omega_M}\frac{d\beta_M}{dt_M}=\frac{1}{\omega_P}\frac{d\beta_P}{dt_P} \tag{3-59}$$

By re-arranging the equation, and introducing new variables, including Ω (named as the effect parameter)

$$\tilde{\Omega}=\omega\tau_S\,; \qquad \tilde{\beta}=\beta; \qquad \tilde{t}=\frac{t}{\tau_S}; \qquad \tilde{\tau}=\frac{\tau}{\tau_S} \qquad \frac{d\beta_k}{d\tilde{t}_k}=\tilde{\Omega}_k$$

$$\tilde{\Omega}_k=\sum_{i=1}^{n}\omega_{ik}\tau_{Sk}=\sum_{i=1}^{n}\tilde{\Omega}_{ik} \tag{3-60}$$

Ω can be viewed as the combination of all transfer processes (or phenomena) with different characteristic times, τsk. To satisfy the covariance principle, the ratio of β and ω between the model and prototype can be a combination of arbitrary constants, λA and λB. By making λA the ratio of all Ωk for the prototype and the model, the scaling criteria (similarity criteria) can be derived.

$$\beta_M=\lambda_A\beta_P\,, \quad \omega_M=\lambda_B\omega_P \tag{3-61}$$

$$\lambda_A=\frac{\tilde{\Omega}_{1,M}}{\tilde{\Omega}_{1,P}}\,;\;\; \lambda_A=\frac{\tilde{\Omega}_{2,M}}{\tilde{\Omega}_{2,P}}\,;\;\; \lambda_A=\frac{\tilde{\Omega}_{3,M}}{\tilde{\Omega}_{3,P}}\,;\ldots\lambda_A=\frac{\tilde{\Omega}_{n,M}}{\tilde{\Omega}_{n,P}} \tag{3-62}$$

With a combination of λA and λB (2-parameter transform), one can derive five different types of scaling (β and ω Coordinate Transformations) shown in the Table 3-5. In reality, some existing scaling methods fit into these categories. For instance, the volume-to-power-scaling belongs to the Identity scaling and the reduced height scaling of APEX test facility falls in the ω–Strain category.

Table 3-5 – DSS: different types of coordinate transformation.

Basis for Process Space-time Coordinate Scaling				
Metric Invariance	$d\tilde{\tau}_P = d\tilde{\tau}_M$	And	Covariance Principle	$\frac{1}{\omega_P}\frac{d\beta_P}{dt_P} = \frac{1}{\omega_M}\frac{d\beta_M}{dt_M}$
$\beta-\omega$ Coordinate Transformations				
2-2 Affine $\beta_R = \lambda_A;\ \omega_R = \lambda_B$	Dilation $\beta_R = \lambda;\ \omega_R = \lambda$	β- Strain $\beta_R = \lambda;\ \omega_R = 1$	ω- Strain $\beta_R = 1;\ \omega_R = \lambda$	Identity $\beta_R = 1;\ \omega_R = 1$
Similarity Criteria				
$\tilde{\Omega}_R = \lambda_A$ $\tau_R = t_R = \frac{\lambda_A}{\lambda_B}$	$\tilde{\Omega}_R = \lambda$ $\tau_R = t_R = 1$	$\tilde{\Omega}_R = \lambda$ $\tau_R = t_R = \lambda$	$\tilde{\Omega}_R = 1$ $\tau_R = t_R = \frac{1}{\lambda}$	$\tilde{\Omega}_R = 1$ $\tau_R = t_R = 1$

This generalized approach also provides the benefit of identifying the distortion at any moment of the transient. By defining a difference variable, η, the following analytical relationship in the (β, Ω) coordinate can be derived provided that the surfaces containing both geodesics are assumed to be flat, e.g. Fig. 3-3. This relationship can be used to estimate the difference quantitatively, i.e. the scaling distortion of the two processes:

$$d\eta_\chi^2 = d\beta^2 + d\tilde{\Omega}^2 \qquad d\eta_\chi = \sqrt{1+\tilde{\tau}^2}\, d\tilde{\Omega} \tag{3-63}$$

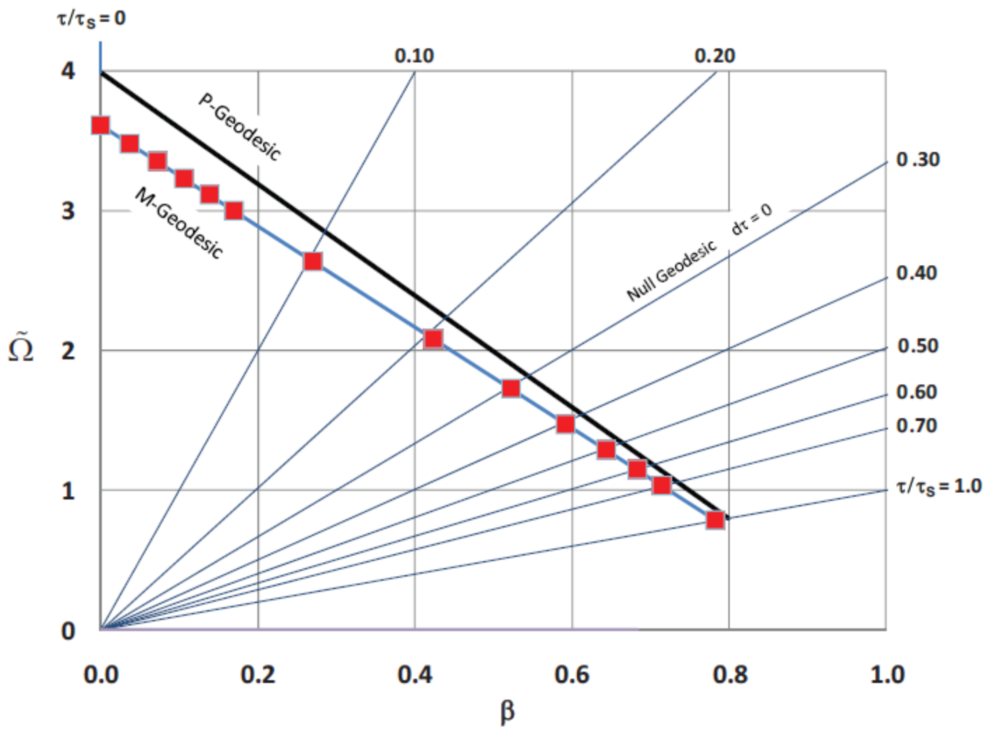


Fig. 3-3 – Process for the identification of scaling distortions in DSS.

Since the DSS approach is a fairly new one only limited numbers of its applications have been published, Yurko et al., 2015, and Frepoli et al., 2015. One may suspect the implementation of the method in a complex reactor-transient is difficult since the analytical derivation is based upon fundamentals which have a general validity. Also the assumption that the geometric phase lines of the model and the prototype are flat in the process coordinate is too optimistic. It is hard to validate that assumption and the calculation of distortion. However, this is a promising approach to describing the scaling distortion quantitatively over the transient period of interest.

Comparison among H2TS, FSA, and DSS

The balance equation of DSS after coordinate transform, eq. (3-53), are similar to the dimensionless ones in H2TS and FSAin terms of their physical meaning, i.e. the rate of change of the conserved quantity of interest is equal to the action ω in DSS. The action is the normalized sum of the Fractional Rate of Change (FRC) in H2TS andFSA. Therefore, the balance equations of DSS, H2TS, and FSA can be re-arranged into a similar form for comparison, Reyes et al., 2015. One difference observed between H2TS and FSAis the normalization process of the agent-of-change. Individual reference values are used for each agent-of-change in H2TS, and an aggregate (sum) of individual reference values is used in FSA. Another difference is the normalization of reference time. In H2TS, the reference time-constant is defined as the largest FRC, and the one in the FSA is the effective FRC. Both time constants are static, viz., they were evaluated at fixed reference values. However, in DSS, the normalized time constant is evaluated in eq. (3-58), which is an integral quantity of temporal displacement-rate over the period of interest in a transient. In terms of assessing scaling distortion, the distortion factor for each agent-of-change formulated in H2TS and FSA is expressed as the similarity parameter ($\prod$ group) deviation of the prototype and the model, divided by the similarity parameter in the prototype. As in the normalized reference constant, they are evaluated at fixed reference values that are outperformed by the time-dependent DSS method described above.

It is worthwhile to emphasize three key differences between H2TS, FSA, and DSS.

Firstly, DSS is a geometry-based method of scaling. By transforming the conserved quantities and agents of change to a phase space (typical of that used in the study of dynamical systems), DSS implements the rules of geometric similarity to establish process similarity. Based on an affine transformation of the $\beta-\omega$ co-ordinates, the generalized DSS method can be transformed into five different scaling-methods including the commonly used volume scaling. Neither H2TS nor FSA have this unique property.

Secondly, the DSS and FSA quantify scale-distortion differently. The FSA determines scale distortion quantitatively from the ratios of fractional rate changes for each specific agent of change. H2TS computes scale distortion by taking ratios of dimensionless characteristic-time ratios. The FSA and H2TS scaling distortions are static. The distortion in different times of a transient can be approximated only by computing the dimensionless parameters through the data evaluated at various snapshots of the transient. Each snapshot would have a unique normalizing factor, and thus, it is not straightforward to compare different snapshots or to integrate the distortion through time. DSS, however, computes the scaling distortion as a function of dimensionless process time via the (flat-space) separation between the geodesics of the prototype and the test facility. The same normalizing factor, the process action, is applied to generate dimensionless quantities, including effective parameter, conserved quantity, clock time, and process time. This allows our comparing the trajectories of the prototype and the test facility process curve for the entire transient as a time-dependent quantity. It may increase and/or decrease as the transient evolves. The geodesic separation then can be integrated to yield a single measure that accounts for all distortion present between the two facilities, or between a code calculation and experimental results.

Thirdly, DSS can make use of a simplified transient analysis model for optimization of an experiment over the entire transient process, Yurko et al., 2015.

3.1.3 Scaling approach

For the design and construction of a test facility, the test objectives and transient scenarios to be simulated should be specified as a first step. All the relevant- and necessary-processes and phenomena thus should be identified during and even before such design work, based on engineering judgement, code evaluation, and/or experiences. The following steps are explained mostly for ITF, but also can be applied for SETF.

One of key factors involved in the efforts of facility design, construction, and operation is the cost (and space) of the facility constructed in a certain size of facility building. A relatively large one, e.g. full height, with BC and IC (boundary and initial conditions) closer to those for prototype, e.g. full pressure and temperature, may cost significantly more in construction, operation, and maintenance. Significant time and money may be needed to develop, install, and operate special yet appropriate instrumentation suitable for prototype test conditions, while this factor does not constitute a technical aspect; it is pointed out here as one of major aspects in the effort on facility scaling.

A Phenomena Identification and Ranking Table (PIRT) then are generated appropriate to the particular transient scenarios. Geometrical scales, such as height and volumetric ratio, operating conditions such as pressure, mass flux, linear heat rate, and working fluid are determined roughly according to the available resources and cost. Also, the influences of time scale and fluid physical properties are considered. Then, global scaling analyses (or top-down scaling in H2TS) for the overall system behaviour is undertaken to obtain similarity criteria and major scaling ratios. Conventional scaling methods, such as power-to-volume scaling and three-level scaling, are used in the global-scaling analysis. All scaling methods have certain inherent advantages and disadvantages so that an appropriate scaling method should be chosen according to the test objectives and the transient (accident) scenarios.

When appropriate scaling methods are not available for some particular scenarios, new scaling criteria are derived from the governing equations and given models/correlations that may well predict the phenomena of interest, however, sometimes the range of test conditions from which the models/correlations are obtained does not match the intended application. The situation is similar to the use of correlations in the code. All scaling methods are framework to be used. They are not constrained by models/correlations. Frameworks are flexible and depend on the users' knowledge Global scaling provides the major scaling factors, such as core power, flow rate, and pressure drop with a particular scaling ratio. A scaling of local phenomena (or bottom-up scaling in H2TS) is undertaken to preserve the important local phenomena in the scaled test facility. If there is a conflict in the similarity criteria between global scaling and local scaling, the priority of the similarity criteria should be established. Conflicts of similarity criteria among local phenomena in a component also need compromises to preserve important- and dominant-local phenomena.

A scoping analysis, using system-analysis codes, is carried out to assess the similarity of the overall system behaviour and major local phenomena in the test facility after the major scale ratios are determined and also the scientific design of the test facility. From the comparison of the system behaviour and major local phenomena, it is possible to identify scaling distortions, and to minimize them by optimizing the design of the test facility. Detailed descriptions of the scoping analysis to determine the scale ratio or to verify the scaling law are given in Section 4.1.5. An engineering design follows the scientific design to meet regulatory requirements, manufacturability, and facility operability. Detailed descriptions on the design and operation of the test facility are given in Section 2.3.5.

In the present Section, the characteristics of the height, time and pressure scaling are described for the preliminary basic design of the test facility. Criteria of the minimum scale ratio related to flow regimes also are discussed (see also Section 3.2.1).

Height scaling

As for full-height scaling (simulation), the analysis of the experimental data and the application to the prototype is relatively simple. Scaling analysis often can be simplified because scaling applies only to the cross-Sectional area of the components, Levin & MacPherson, 1995, and Boyack et al., 1989. When full-height scaling is adopted with power-to-volume scaling, the time scales are preserved in the scaled-down facility. The preservation of the time scale is very important for fast transients, such as an LB LOCA. The full-height conservation is necessary for correctly representing natural circulation driving force, especially through the primary coolant circuit in both BWR- and PWR-simulations. The power-to-volume scaling also simplifies the analysis of data.

However, in small (thus thin, slender) facilities, there are inherent deficiencies compared to the full-height facility designed with power-to-volume scaling. Multi-dimensional thermal-hydraulic behaviour is limited in representation when the volumetric scale is very small, because the hydraulic diameter of the main components and/or the piping may become too small. The observed phenomena may then become rather one-dimensional. Simulation of parallel channel-flow, such as in U-tubes in PWR steam generator (SG) is limited because the number of U-tubes is very few. In addition, distortions of heat sources and sinks can be significant, relative to the prototype. The influences of excessive heat-transfer to the fluid, excessive stored heat and/or excessive heat loss in metal may affect the thermal-hydraulic response, such as local distribution of steam generation, and thus, the progress of the transients.

Another problem is the preservation of pressure drops in the scaled-down test facility. Hydraulic resistance, especially in the loop pipes, rises significantly when the pipe diameter is greatly reduced. The preservation of, or the reasonable reduction of hydraulic resistances is important for SBLOCAs whose cooling is dominated by natural circulation, Hsu et al., 1990. Generally, the diameters of the loop pipes are oversized, and their horizontal lengths are shortened to assure the correct hydraulic resistance while maintaining the scale of the fluids' volume in the pipe. The diameter and length of horizontal legs then are determined by employing the Froude number, for example, ROSA-IV Group, 1985. Related discussion is given in Section 3.2.3.1. The distortions in heat sources and sinks, coupled with the difficulty of scaling pressure drops with very small diameters, can distort the system natural circulation behaviour.

Non-prototypic effects or phenomena such as the effects of surface tension and transitions in the flow regime are often observed in pipes very small diameters in the test facility.

A facility with reduced height compared to normal facility often is adopted to avoid and minimize the limitations and disadvantages of the full-height facility, Levin & MacPherson, 1995. The scientific basis of this scaling is the balanced simulation of friction and gravity to preserve the single- and two-phase natural-circulation phenomena. For a reduced-height facility, the scaling analyses, analysis of the experimental data, and the application to prototype are complex compared to that for a full-height facility. In particular, complex behaviour of the thermal-hydraulic-flow is dependent on components' heights, including the heterogeneous behaviour of U-tube flow, flow stagnation, or reverse flow in individual tubes; for instance, they may be difficult to replicate realistically. The time scale is not preserved in the reduced-height facility, so that it is difficult to preserve thermal-hydraulic phenomena in a fast transient. Thus, such facilities are appropriate for simulating slow transients, such as SB LOCA and operational transients. The ATLAS adopting reduced-height shows that the natural circulation flow maps, D'Auria et al., 1991, are similar to those of other integral test facility for PWRs, and are well within the envelope for expected natural-circulation situations in typical PWRs, Choi et al., 2014.

Compared with a full-height facility with a large volumetric scaling ratio (a very slender facility), the multi-dimensional phenomena can be well preserved in the reduced-height facility due to its having an aspect ratio close to the prototype, Song, 2006. Scaling distortions, such as those originated by structural heat loss and pressure drops can be decreased in the reduced-height facility compared with a full-height test facility with the same power-to-volume scaling.

In the reduced-height facility, it might be difficult to preserve the local phenomena, such as critical heat- flux (CHF) and reflooding in the reactor core. Both phenomena are very complex, and are strongly affected by local parameters, as well as boundary conditions. The distortions of multi-dimensional phenomena, such as downcomer boiling and ECC bypass, affect the hydrostatic driving-head for core-inlet flow, and thus result in distortions of the CHF and re-flood phenomena in the core. The distortions of the boundary conditions will have much more significant influences on the CHF and the reflood phenomena compared with inherent scaling distortions due to the reduced-height.

The reduced-height scales often vary from a half- to a quarter-height scales. Discussion still continues on an optimal height scale for a reduced-height facility. In addition, there are no criteria on the minimum-height scales without significant scaling distortions. The optimal height scale may be chosen considering various aspects, such as processes and/or phenomena to be simulated, experimental objectives, limiting phenomena with severe scaling-distortions, and so on. Kocamustafaogullari & Ishii, 1984, applied the scaling criteria to a conceptual design of an integral test facility for simulating a PWR with once-through steam generators. Their calculations showed that the most severe similarity limitation was imposed by the similarity of the frictional pressure drop over the hot leg. They provided an optimal length-scale as a function of the areal scale from the preservation of the similarity criteria of the frictional pressure drop over the hot leg. It should be noted that the results are not general for choosing the optimal height scale, but depend upon the prototype of interest. Sometimes, a scoping analysis using numerical codes is helpful to determine an optimal height-scale for an integral test facility, Ransom et al., 1998, and Park et al., 2007.

Time scaling

Time scales with the power-to-volume scaling are preserved in the full-height test facility, opening up the possibility of attaining the same timing of events and local thermal-hydraulic responses, and thus to reproduce the mass/energy distribution and heat-transfer responses of prototype. Meanwhile, the time scales associated with the reduced-height facility are usually reduced (accelerated or shortened) in the test facility relative to the prototype. There can be advantages and disadvantages of the time reduction scaling.

As also discussed above, time preservation is of primary importance for fast transient scenarios, such as LBLOCA. The time-accelerated scale in the reduced-height facility may cause some complexities in its operation and control during a fast transient experiment. The reduction in time may be appropriate for simulating slow transients, such as SBLOCA, and operational transients. Especially, the acceleration of time can be useful for simulating long-term behaviour, such as in passive systems, Peterson et al., 1998.

There are phenomena with their own time-scales, such as vapor generation and condensation that may be distorted. Scaling methods using validated models of both processes might, in principle, account for these time scales.

Pressure scaling

Pressure scaling is closely related to quality and uncertainty evaluation of simulated thermal-hydraulic phenomena because the properties of fluids change in terms of their pressure and temperature. Full-pressure simulation of prototype phenomena, with considering scaling, thus is the method mostly employed, especially in ITFs. Replication of prototype phenomena may be possible because of the same fluid properties, provided that it is not necessary to consider the influences of radiation.

Reduced-pressure facilities also have been used with an appropriate pressure scaling when some constraints exist in the design, construction, and operation of facility. However, test facilities operated under reduced pressure (with an upper limit on the operating pressure) that correspond exactly to the prototype pressure do not need to consider pressure scaling. PKL can be classified into this category, Umminger et al., 2012, and Umminger et al., 2013. The facility is designed to represent phenomena under full-pressure conditions, but is constructed to operate within a reduced pressure range (5 MPa) relative to the prototype. Investigations are performed which include parametric studies on those phenomena and processes expected to entirely evolve in the prototype below the limitation pressure of the ITF; the CCFL during reflux condenser mode (of natural circulation) and transients during cold shutdown conditions are typical examples. When the pressure of an accident or operational-transients exceeds the ITF limitation pressure, phenomena at the higher pressure cannot be reproduced. Nevertheless, in many cases, relevant phenomena can be reproduced qualitatively under lower pressures. In many experiments, the main part of the transient starts at a specific entry-point around the ITF limitation pressure and the rest of the transient that includes the main phenomena of interest (e.g. the restart of natural circulation, ACC injections, or the influence of inert gases on heat removal), evolves at the pressure range expected for the prototype. The BICs for the entry-point are provided by TH SYS code analyses, and/or by full pressure ITF experiments in the frame of counterpart testing (Section 3.3). Water is used as working fluid, equal to the prototype in this category. Reduced structural masses may help to minimize distortions in the phenomena and the time constants.

UPTF, CCTF and SCTF used for 2D-3-D Program, Weiss et al., 1986, USNRC, 1988, Mayinger et al., 1993, and Wolfert, 2008, although these are SETFs, can also be classified into the category of ITF. The test objectives mostly were to observe refill-reflood phenomena during a PWR LBLOCA that occur within narrow pressure range lower than around 2 MPa. The observed phenomena may be dealt with equal to those observed in the reference reactors.

Pressure scaling is necessary for ITFs designed for operating under reduced-pressures, to simulate thermal-hydraulic phenomena that may occur in full or some selected pressure-range in the reference reactor. The experiment is performed under reduced pressure and in a small range that proportionally corresponds to the high- and wide-pressure range in the prototype. Similarity parameters are preserved in reduced-pressure experiments the same as those in the prototype pressure. Some experiments can be started from the beginning of the accident to be simulated. The BICs are determined by scaling analysis. The SYS TH code analyses are sometimes used to support the scaling analysis. Either water or a simulant (a non-prototypical fluid. such as Freon) is used as working fluid, depending on the objectives, method, and cost of the experiments, although it is difficult to use both fluids alternatively in one facility. Thus, this category can thus be divided into two, depending on the working fluid, as follows:

(1) Water as working fluid: Detailed scaling analysis is given by Reyes et al., 1998, based on H2TS, indicating a good linearity in the pressure response of a reduced-pressure ITF and the reference reactor for depressurization transient during an accident, over a limited pressure range of interest. There are great merits to conducting experiments under reduced pressures for operation, maintenance, and instrumentation. No specific problems arise in using the SYS TH codes to support the estimation of BICs. Typical ITFs that adopt this type of reduced-pressure scaling are APEX employed to simulate the AP-600, AP-1000 ,and CE Palisades reactors, e.g. Reyes & Hochreiter, 1998, Reyes et al., 1999, Welter et al., 2005, Reyes, 2000, and Reyes, 2001, and UMCP and SRl-2 were used to simulate a B & W reactor, Larson, 1987. Difficulties in this type of reduced-pressure scaling were recognized, e.g. Kocamustafaogullari & Ishii, 1987, Larson, 1987, Hsu et al., 1990, and Di Marzo et al., 1991. A large scaling distortion in power or time also may appear, being caused by discontinuity in the phenomena simulation between single-phase liquid flow and two-phase flow, e.g. Larson, 1987, and Kocamustafaogullari & Ishii, 1987. Finally, Larson, 1987, points out that a change in power level may be necessary when a single-phase liquid flow turns into two-phase flows in natural circulation simulation, for instance.

(2) Simulant as working fluid: This is called a fluid-to-fluid simulation, which enables the utilization of simulant fluid (a non-prototypical fluid), Kocamustafaogullari & Ishii., 1987a. Refrigerants, such as R-11, R-113, and R-134, have been used to simulate various two-phase flow phenomena, such as the critical heat flux, and boiling. A typical ITF is DESIRE that simulated BWR and used Freon 12, De Kruijf et al., 2003. Cost savings in the facility construction and experiments would be possible because of lower pressure and temperature in

the experiments. Some special preparation may be necessary to prevent leaking from the facility. High-temperature conditions, such as boiling and temperature excursions in the simulated core, may cause the pyrolysis of the refrigerants. Related discussions are given in Sections 3.2.3.1 and 3.2.3.2.

Number of loop scaling

The reduction of the prototype PWR number of loops (including all interfacing safety- and operational-systems) allows a considerable simplification of the ITF, and thus the possibility for reducing the cost for both facility construction and operation. A preservation of the prototype number of loops generally is deemed favorable, especially in the cases where presumed asymmetries in BIC among the individual loops, and their consequences on the overall system behaviour play an important role. In most accident scenarios, the initiating event, e.g. a SB-LOCA, SGTR, or a MSLB, occurs typically in one individual loop. Overlapping with additional single failures, e.g. the unavailability of safety systems or partly isolated SGs, such an accident scenario can lead to very heterogeneous/asymmetric behaviour with a different BIC for all of the three or four loops.

Full loop ITF, additionally equipped with all relevant interfacing systems on the primary side and on the secondary side, will allow the investigation of a broad spectrum of accident transients, comprising a variety of all feasible BIC in different loops. In particular, a full prototypic number of loops allow a better replication of BIC for natural circulation under asymmetric conditions, e.g. impact of counter-drive for natural circulation in loops with isolated SG following SGTR versus 2 or 3 loops with intact SGs. The stagnation of natural circulation in a loop with an isolated SG (e.g. after a SGTR) determines the possibility of reboration of the affected loop and is an important aspect in assessing the risk of re-criticality. The tendency for flow stagnation depends significantly on the ratio between the affected loop and the intact ones. The reproduction of this different behaviour in the individual loops is limited in the case of lumped loops.

This also is applicable for the BIC for coolant mixing in the RPV downcomer wherein the coolant flows from different loops, arriving with different flow rates and temperatures, and is the basis for mixing effects in the down comer, e.g. with respect to re-criticality issues for inherent boron-dilution following an SB-LOCA, or a PTS following an MSLB.

Some ITFs are equipped with a lumped loop with different loop diameters from single loop, e.g. one loop representing three intact loops, and one loop representing the broken loop. In this case it is not possible to adjust either the elevation of the centerline, or the bottom or the top of these two loops to the same level, which may affect the break-flow upstream liquid level. In general, lumped loops are sensitive to mass distribution and may influence the uncovering phenomena of the core during an SB-LOCA.

On the other hand, the combination of several loops into one, the so-called ‘lumped’ loop, yields larger diameters and a smaller surface-to-volume ratio for the RCS piping system. Consequently, both are closer to the prototype PWR, and thus provide a better replication of 3-D phenomena, such as coolant mixing, thermal stratification, and counter-current flows in horizontal legs, and lesser heat losses compared to a full-loop ITF stringently scaled to the same volume-scaling factor. For a small-volume-scaled full-height facility, loop lumping has a merit in decreasing the resistance to flow by increasing the pipe diameter.

The preservation of those phenomena relevant to transport characteristics also is an important aspect for both types of ITF, and is usually realized by enlarging the pipe diameter (deviating from rigorous volume scaling) and preserving the Froude number (e.g. PKL with four loops, and BETHSY with three loops).

Criteria for minimum-scale model dimensions

One of the criteria that should be taken account for in the scaled-down facility is to maintain the facility dimensions enough to preclude effects due to size that would not be expected to occur in the prototype, Boucher et al., 1990. For example, if the surface tension does not affect phenomena at the prototype, then the dimension of the facility also should be large enough to preclude the effects of surface tension on the phenomena. Boucher et al., 1990, established the concept of a minimum dimension from flooding considerations. The criterion was based the following dimensionless diameter:

$$D^* = [gD^2\Delta\rho/\sigma]^{1/2} = \sqrt{Bo} \quad (3\text{-}64)$$

As long as the dimensionless diameter is greater than approximately 32-40, the geometry can be considered to be large with respect to surface tension influences. This criterion can be used to determine the minimum size at the scaled-down facility. It usually serves to determine the size of the reactor vessel downcomer in the scaled-down facility. Other criteria for the minimum-scale dimensions can be derived considering the preservation of hydraulic resistance (friction numbers), stored heat, heat losses, flow regimes, and so on, relative to the prototype.

Levy, 1999, observed that if the facilities are at least one third in the size of the prototype, there will be minimum impact of the scale or size.

Role of system-analysis codes in the scaling approach

The role of system codes in scaling is discussed into detail in Chapter 4. System codes are widely used to analyse experimental data, and even to support the results of scaling methods in designing facilities and experiments. Therefore, few notes about the role of codes are given hereafter (Chapter 4 gives a broader coverage of this topic).

After deriving scaling criteria by the scaling laws, and completing the basic design of the test facility, it is possible to assess scaling distortions of system behaviour in the test facility by using similarity parameters. Scaling of transient can be undertaken because the transient terms are included in the scaling method. The transients usually are divided in phases (periods), and the distortion is assessed for each period. The scaling distortions of important local thermal-hydraulic behaviour in each period can be evaluated for each component based on those properties evaluated at the time of interest. However, to assess the scaling distortions on transient behaviours, it is easier to perform scoping calculations using the system-analysis codes to investigate the similarity of transient system behaviours under postulated BICs (boundary and initial conditions).

For this purpose, in the process of scaling and in the preliminary design of a scaled-down facility, system codes or CFD codes are used to investigate the BICs of planned experiments based on the preliminary design of the test facility. The system code models usually are created for both the prototype and the scaled-down facility in order to compare their predicted responses, Ransom et al., 1998, and Reyes et al., 1998. The main objective there is to ascertain whether the scaled-down facility, which is scaled down from the first principles of scaling laws, would exhibit similar system behaviour to particular transient expected to occur in the prototype, Reyes et al., 1998. The comparison of the code calculation between the scaled-down facility and the prototype is useful to preliminarily confirm that important processes are identified and addressed in the scaled-down facility within the range of past knowledge on the reactor response. In addition, the comparison may help to identify origins of the scaling distortion in the test facility, and to understanding the effects of the scaling distortion as well as the facility biases due to the scaling distortions. The scoping calculation thus is helpful in minimizing the scaling distortions in the test facility mainly through the experimental BICs. The data so obtained are used to validate the system codes that were employed for such preliminary evaluations on the experimental BICs.

As pointed out by Levin et al., 1995, scoping calculations may be undertaken with code models that have not been validated for their application to the prototype or the test facility. This could lead a fundamental mistake should the non-validated codes be used to identify BICs for experiments that will, in turn, be used to assess the codes. To avoid this, it is important, at both the scaling and design stages, that the codes are used in a limited fashion, simply as one means of assessing aspects of the design. The results of scoping analyses should not be used to establish the design of the test facility; rather, this should be done by applying the key scaling parameters derived through an independent scaling analysis. Levin et al., 1995, further detailed the use of computer codes as scaling- and analysis-tools.

3.2 Scaling and experiments

Through the history of design, safety assessment, utilization, and maintenance of the LWRs, thermal-hydraulics has played a center role, especially in fluid dynamics, heat transfer, and in developing the nuclear core. Appendix 2 gives an outline of the history of nuclear thermal-hydraulics. Experiments then were utilized, even before the advent of computers, to estimate, understand, and prepare models of thermal-hydraulic phenomena that may appear in the prototype LWRs of various sizes and the initial & boundary conditions. Therefore, experiments have formed the basis of nuclear-thermal-hydraulics to meet requirements of the safety evaluation of LWRs, being connected to the development of computational tools that include the SYS TH codes. Currently, a tight connection is established between experiments, development and the qualification/validation of computer codes as described in Chapter 4.

Experiments can be classified into three categories, viz., basic tests, separate effects tests (SETs) and, integral tests or integral effect tests (IETs), NEA/CSNI, 1993, NEA/CSNI, 1996, and USNRC, 1988. Here the word “test“ is used with the same meaning as experiment. These categories of experiments briefly are explained here to provide some introductory statements related to the contents of Section 3.2.

<u>Basic tests</u>

This category of experiments addresses fundamental phenomena, such as pressure drops, single- and two-phase flows, fluid mixing, heat transfer, including boiling and condensation, critical flow, pressure- wave propagation, and complex phenomena due to combination of fundamental processes like flooding and countercurrent-flow limitation. Basic tests aim at understanding the phenomena mostly under simple and steady boundary conditions, sometimes with less reference to actual LWR conditions, including those expected in accidents. Rather basic tests may reveal information essential for developing models and correlations also embedded into balance equations that are part of the SYS TH codes. Since basic tests have a weak connection with scaling as noted above, they are not mainly dealt with in the following sub-Sections. However, their essence is discussed in Section 3.2.1.

<u>Separate Effect Test (SET)</u>

Validation of codes and models should become practical when local phenomena are separated from the system response where various phenomena interact. A reasonable list of separate effect phenomena was thus established, NEA/CSNI, 1993. For example, local phenomena suitable for SET are expected to occur in and around the following: a) Primary thermal-hydraulic regions or zones of LWRs, b) components, such as centrifugal pumps, valves, separator, dryers, jet-pumps, accumulators, and also c0 control rod guide tubes, during specific phenomenological windows that can be identified in accident analysis such as “refill” with 3-D fluid mixing, and “reflood” with 3-D & non-equilibrium quench front behaviour. Scaling is important for the SET because attention can be devoted to certain local phenomenon (within suitable range of variations of the key parameters) although it is a part of the system performance. In the following sub Sections, 3.2.1 deals with methods on facility design, and the setup of experimental conditions related to SETs, while Sections 3.2.2 and 3.2.4 describe and discuss key characteristics of the facilities respectively utilized for SETs on reactor systems and the containment, to provide the necessary information to discuss their influences on scaling.

Integral Tests or Integral Effect Tests (IETs)

Integral test facilities (ITFs) have been designed and operated to try to reproduce the reference- reactor performances that were anticipated from the best-available SYS TH code analyses and/or by various scaling analyses. However, unavoidable distortions may appear and prevent the achievement of this goal, as discussed in Chapter 2 and in Section 3.1. A key goal of the IETs then is to provide data to validate the predictive capabilities of the SYS TH codes as discussed in Chapter 4. Scaling undoubtedly is essential and decisive for designing and the operating ITFs by considering best attributes of, and the limitations in related projects. The sub Sections 3.2.1 also deals with methods of designing facilities and setting up experiment conditions for IETs; Sections 3.2.3 and 3.2.5 describe and discuss key the respective characteristics of facilities utilized for IETs on reactor systems and the containment.

Nuclear Power Plant (NPP) data

Transient data are measured with very limited instrumentation in NPPs during operational tests, e.g. commissioning, start-up, and various unplanned situations, including accidents. Although they are typically not suited for code assessment to the same extent as are basic experiments, SETF and ITF. However, NPP data do not suffer of any scaling limitations. NPP data are suitable for comparison with code results, e.g. Hirano & Watanabe(1992), and Reventos et al., (2008), and already have been adopted within the framework of code application/validation by international institutions, e.g. NEA/CSNI – ISP 20, 1988, NEA/NSC, 2001, NEA/NSC, 2001a, NEA/NSC, 2002, (see also NEA/NSC, 2006), and, NEA/NSC, 2009, based on data measured in NPP units, viz., Doel-2, Peach Bottom, Kozloduy-5, and Kalinin, respectively. These are performed under a certain common understanding such as: (1) most of the data are proprietary, especially in relation to geometrical parameters: computer code input was provided by NPP owners in the case of the Doel-2 SGTR case, for example; (2) the data provided usually is quite limited in relation to the accident of interest, e.g. Mihama Unit-2 SGTR, Hirano & Watanabe, 1992.

Owing to the reasons mentioned above, the mostly narrow range of variations for measured parameter and the limited number of instruments suitable for characterizing transient scenarios, NPP data are not considered further in this document. Nevertheless, it is recognized that they may contribute to validating system code models and/or approaches to developing nodalization.

Scaling Distortion

It is difficult to eliminate scaling distortions in either SET or IET because of the many limitations in the design and operation of facilities, which undoubtedly prevent the achievement of the ultimate goal, i.e. to reproduce the expected phenomena in the reference reactor. Section 3.2.6 discusses the influences of several limitations in the experiments.

3.2.1 Facility design and establishment of experiment conditions

Roles and requirements for experiments (data)

In the processes of development and safety evaluation of nuclear reactors, thermal-hydraulic analysis of the reactor coolant system (RCS), the containment system, and their coupling is essential in understanding operational and/or transient phenomena that may happen in the reactor design of concern (e.g. PWR, VVER, BWR, and, CANDU). This analysis starts with the development and assessment of an experimental database to characterize the possible prototype system thermo-hydraulic behaviours as correctly as possible. Computer codes then can be developed and validated against this database, and used for safety analysis of the reactor.

Since it is difficult to readily construct a full-pressure and full-size test facility to obtain such assessment data, and to perform detailed measurements for all the required parameters necessary to develop and verify thermal-hydraulic models, reduced-scaled test facilities; i.e. integral test facility (ITF) and separate effect test facility (SETF) often are used instead. The scaling compromise then arises during the design process of the scaled test facilities and the understanding of the experimental results obtained from them because it is difficult to apply all of the scaling factors specified, from the scaling methods into the facility design. The scaling distortion then inevitably occurs in any of scaled tests.

Identification of the safety margin in important parameters such as peak clad temperature or containment pressure, for specific reactor designs in case of accident and abnormal transients is the basic role of safety analysis. The experimental data obtained through ITF- and SETF-experiments thus should address this main purpose through validating the system codes that were developed by using the data also obtained from other the ITF- and SETF-experiments. Since there is limited amount of data, and none in most cases, for the reference reactor, conservatism in input data is needed for safety analysis and correspondent suitable conservatism is expected (not easy to demonstrate) in the results.

Throughout the history of reactor safety research on reactor accidents and transients, as well as experiences from the ITF experiments that simulated transient thermal-hydraulic responses during accidents and abnormal transients, important phenomena have been recognized and requirements for experiments thus have been summarized in a form of phenomena identification and a ranking table (PIRT), see NEA/CSNI, 1993, NEA/CSNI, 1996, and USNRC, 1989.

The OECD NEA established the NEA Data Bank to gather and provide experimental data, mostly from the scaled facilities necessary for developing and validating system codes for safety analyses. There are several other databases that are difficult to access because of specific ownerships by research institutes, international agency/union, and even government. Most important phenomena that may be encountered during reactor accidents, however, are covered in the internationally available database for validating system codes, such as the NEA Data Bank.

Separate Effect Test Facility (SETF)

The major role of SETF is to provide experimental data to develop and validate physical models and/or empirical correlations under prototypical- or simulated-conditions. The former condition corresponds to the reproduction of phenomena that may appear in reference reactor. In many cases, the SETF is designed for experiments under steady-state conditions, because it is convenient to develop and validate physical models and empirical correlations. Depending on the phenomena of interest, SETF is used also for simulating and reproducing a dynamic transient.

SETF is designed to reproduce the thermal-hydraulic phenomena expected by system code analyses and/or other experiments, thus is based on experiences. A good set of measurement instrumentations is planned that are being furnished with as many parameters with as high spatial resolution as possible. In many cases, however, it is not easy to readily define the design of certain SETF enough to cover the required- and expected-phenomena because the LWR is operated under conditions of high-temperature and high-pressure. To envisage physical models of phenomena in a somewhat step-wise manner, and to consider and develop instrumentation to measure required but difficult parameters for developing and verifying the physical model, air-water experiments are the first choice in many cases, so to prepare for steam-water facilities in the next step.

System code with physical models and empirical correlations from SETF experiments usually are used to identify thermal-hydraulic responses by simulating various types of accidents and abnormal transient of reactors of interest. The experimental conditions for SEFT (boundary conditions [BC] and initial conditions [IC]) then are planned to include all the conditions expected by the system code reactor analyses, so that the expected uncertainty in the results from it are covered within the range of SETF operation that controls the BC and IC. During the design process of SETF, which is being coupled with the

BC and IC considerations, the scaling approach and scaling methods for each phenomena of interest, dealt with in Section 3.1, are considered through the code analyses to eliminate or minimize scaling distortions.

As long as the BC and IC are well defined in a SETF, one may expect the similar conditions of the local phenomena based on the results from other SETF(s), even though the geometry of other SETF(s) is not exactly the same. However, care should be necessary in making the comparison because the evolution of thermal-hydraulic phenomena should take place with a certain length in the flow path. The resulting local phenomena can be different from each other even though the local conditions expected by a given physical model are similar, Zuber, 1980.

During the reactor accident and the abnormal transient, various components in the reactor system are involved and thermal-hydraulic phenomena in each component may interact mutually in a complicated way throughout the whole transient. Such mutual interactions are not completely understood. Reactor core conditions, for example, such as mass flux, quality, void fraction, flow orientation, power profile and even channel shape, may dynamically change according to the flow and pressure conditions in other portions of the reactor system. Three-dimensional non-uniform flow, sometimes in counter-current conditions, may arise partially depending on the reactor accident scenarios. SETF then is designed and used to simulate and reproduce a limited range of phenomena within the expected local phenomena during a certain type of reactor accident and/or abnormal transient, because it is difficult to adapt a single test-facility into all types of thermal-hydraulic responses. The SETF is designed to have auxiliary system, such as the water-fill system with its heater and pressurizer to control the coolant condition as intended. When experiments are performed, BC and IC are controlled within the intended conditions by avoiding influences from other systems, even when such influences may be inevitable during the accident of interest.

PIRT is one of the important guides in identifying and defining the objectives of the SETF experiments. Once these objectives are defined, the SETF specifically is designed and used either for local, steady and average data, such as critical flow, or rather integral, component-wide and dynamic data such as core re-flooding during a PWR large break LOCA (LBLOCA), see NEA/CSNI, 1993, NEA/CSNI, 1996, and USNRC, 1989.

The type of SETF is either for a steady-state condition or for transitional dynamic condition of the phenomena of interest. It sometimes is difficult to distinguish the SETF from ITF, if the transitional dynamic condition of the phenomenon is concerned because the auxiliary system for such SETF becomes rather similar to the components of reactor. UPTF, CCTF, and SCTF used for 2-D-3-D Programs in the United States, Germany, and Japan may fit the SETF, as their data is limited to a certain portion of the PWR LBLOCA scenario. Their BC and IC then were carefully defined to be well adapted to the required conditions of interest that should cover the expected range of the transient. Some details on the CCFL are given in Chapter 2 (Sections 2.1.3.1, 2.1.3.2, and 2.3.2), Section 3.1, and Chapter 4 (Sections 4.3.4.1 and 4.3.4.2).

Consequently, numerous kinds, and thus an enormous number of SETFs have been performed, corresponding to various aspects of local phenomena of interest, so to develop and validate the corresponding physical models and empirical correlations in the codes. Since each of these models and correlations, for developing the system code, considers local single phenomenon that may not always scale to the reactor. Therefore, there are an enormous number of physical models and empirical correlations with limited applicability within a small range of conditions. Extrapolation such results from SETFs are considered afterwards, by reviewing the system-code calculations, other SETF- and even ITF-results. Actually, since the closure laws in the system code mainly are based on scaled SETF experiment data, extrapolating code results remains as open issue. While the range of experiment conditions of each SETF is only partial compared to the required range for analysing reactor phenomena, aggregating many overlapping results may cover a certain range of phenomena, possibly available for their extrapolation to reactor conditions once the extreme end of applicability range will include the prototypic conditions.

Most phenomena are under influences of conduit geometry, which may include the flow pattern in horizontal pipes, including the conditions of flow that are not fully developed, and the counter-current flow of liquid and steam in the bend in the HL of PWR. Thus, directly extrapolating the measured data to full scale data for validation is not possible. Therefore, for predicting such phenomena with no accessible experimental evidence, a scaling analysis is undertaken to estimate the needed capabilities for correlations and models with the highest achievable precision. Uncertainty analysis is performed further when safety margins are evaluated (not only) for the cases where the information available from the experiments is incomplete (see also Sections 4.4 and 4.5).

Recently, a heavily instrumented SETF to produce spatially and temporally very-fine resolution data is called a CFD-grade experiment, suitable for validating CFD codes, although the thermal-hydraulic phenomena generally are three-dimensional and inherently dynamic. It then is important to distinguish the steady phenomena from the dynamically changing phenomena because it is not easy to obtain good set of measurement data for the latter case. The importance of uncertainty estimation in the calculated result is equal to that for system code provided that the CFD codes are used for resolving the problem of reactor safety in this sense, the role of the SETF should be equal for developing and validating both the system code and the CFD code. The CFD-grade experimental results are useful for validating the physical models and the empirical correlations for the system codes too.

Four levels of validation are under discussion concerning the applicability of models and correlations obtained under steady-state conditions to transient conditions. The four levels of data correspond to the basic experiment, SET, IET and NPP, from which a narrow ranges of variations for parameters are available. Here, the third level (IET) of validation specifically addresses the issue of demonstrating the capabilities of correlations developed from steady-state experiments and applied to evaluating transient conditions.

Integral Test Facility (ITF)

ITF is a test facility to provide dynamic- and similar-thermal-hydraulic responses that may appear through postulated accidents and/or abnormal transients in the reference reactor. A whole system is simulated with, at least, the heat source and heat sink within a closed loop, so that each corresponds to the major reactor components. Then, ITF has the capability to simulate a whole transient of postulated accidents and/or abnormal transients. Steady-state experiments, such as steady forced-circulation, including the reactor nominal operating conditions and natural circulation, constitute a part of ITF role to understand the fundamental response of reactor design, as well as characteristics of the ITF itself. Since the local thermal-hydraulic phenomena interact among each other, the data obtained are suitable for understanding the thermal-hydraulic response in the whole system, as well as in each of scaled components with their mutual influences.

While it appears that the ITF experiments provide a similar response to those expected in the reference reactor, the data obtained from the ITF experiments are considered not directly applicable to full-scale conditions, in place of a reference reactor. Instead, the data obtained is used mostly for validating system codes, and understanding of accident phenomena, especially their effect on such major parameter as the PCT as a result of liquid level transient in the core during various accident scenarios. While the measurement instrumentation of ITF is not so abundant compared with that for SETFs, the data obtained from ITF experiments sometimes are utilized for validating the CFD code for local phenomena.

When the ITF is designed, first its purposes are considered and then several scaling approaches are considered, e.g. Section 3.1.3, to fit the within practical limitations, such as size of the budget, and size of the building. Major thermo-hydraulic phenomena are considered simultaneously by using system-code analyses on target accidents and abnormal transients. Major design parameters, such as pressure, height, and volumetric scaling of the ITF are defined in such a first step of the ITF design process. Once the purposes are defined, or along with the consideration of the purposes, an appropriate scaling method (see

Section 3.1.2) is selected to design the ITF with major BCs and ICs that are constrained by the major conditions employed, i.e. pressure, height, and volume.

Scaling distortion (see Section 2.2) is inevitable for the ITF and can be the origin of uncertainty in the safety analysis. Therefore, the minimization and/or elimination of the scaling distortion during the facility-design phase along with an estimation of the scaling distortion during the design phase and after completing construction are among the most important tasks relevant for the design process of the ITF.

Since the ITF is a facility to rigorously observe the interaction of phenomena that arise in each major component at the right timing. The scaling distortions especially related to the time advancement should be minimized or be eliminated. This is true because there are many of physical phenomena that we cannot temporally control, such as critical flow at the (pipe) break and bubble rise velocity according to the fluid physical properties and the bubble size.

The definition of BC and IC for the ITF is required to represent the conditions of the whole system including the reactor normal operating condition. Many BCs and ICs cannot be controlled by scaling laws or criteria, from the experiences of SETF- and ITF-operations. Therefore compromises are needed to reduce their effects. Examples are a) simulation of the fuel pin (its structure, materials, etc.), b) heat loss, c) the pump characteristics, d) pressure distribution, and e) the valve operation. In relation to the pump characteristics, homologous curves in scaled pumps are generally different from prototype pump homologous curves. Estimation and compensation for heat loss, in relation to the heat source (core power, pump power, feed-water, and ECCS) and heat sink (SG, break/leakage, steam-line, relief and safety valves) also would also be one of the subjects to consider as a suitable BC to attain appropriate heat balance for the system, especially during a long-term transient. Therefore, it is necessary to evaluate the scaling distortions associated with BC and IC, and to establish proper countermeasures, e.g. NEA/CSNI, 1989, Karwat, 1986, NEA/CSNI, 1993, NEA/CSNI, 1996a, NEA/CSNI, 1996d, NEA/CSNI, 2001, USNRC, 1988, D'Auria & Galassi, 2010, Levy, 1999, and Mascari et al., 2014.

However, some scaling distortions would be acceptable if the experimental results obtained can be successfully scaled-up via the scaling method used to design the ITF, and they compared well with the expected results of the prototype predictions by the given system code. When the experimental results are utilized to validate system codes, as well as the physical models, however, the experimental results obtained can be compared directly with those of the system code in real-time, provided that the calculation model (input) reproduces the ITF, as is, in the code-analysis. This comparison is valid within the scaled ITF conditions.

A portion of ITF can be utilized as SETF to simulate local thermal-hydraulic phenomena when the BC and IC can be well defined within the ITF capability by controlling feed and break conditions. This actually was achieved at UPTF to simulate flows (flow regimes) in the HL and the CL. In various ITFs, tests for response characterization are performed for components like the core, pressurizer, accumulator, pump and valves. In LSTF, void distribution tests were performed in the core to obtain interfacial drag under prototypical high-pressure conditions, e.g. Anoda & Kukita, 1990.

Here it is noted that a practical method to estimate uncertainty (scaling distortion) inherent for each ITF design is the objective of an uncertainty evaluation to account for uncertainty associated with scaling distortions. This is discussed in more detail in Section 4.3 (definition of the problem), and in Sections 4.4 and 4.5 showing current ways to resolve the issue.

Summary on SETF and ITF Design and Establishment of Experiment Conditions

Table 3-6 compares major characteristics of SETF and ITF from their merits and issues, based on the subjects discussed in this Section and those in Section 3.3.

A huge amount of experimental data has been gathered for clarifying nuclear thermal-hydraulic phenomena, validating computational tools, and performing scaling analyses or even validating scaling methods, see Appendix 2: An Outline of the History of System Thermal-Hydraulics.

The major characteristics of the SETF and ITF experiments are summarized in Table 3-7. The key facilities, mainly ITF and selected SETFs, designed, constructed, and operated in the world can be selected from Appendix 3. More details for a few PWR ITF are given by Belsito et al., 1993, and Ingegneri & Choinacki, 1997. The scaling issue for SETF- and ITF- also is specifically addressed in NEA/CSNI, 1996a.

Appendix A3 should be seen as a by-product of the S-SOAR activities, and will not substitute for the activity recommended (Chapter 5) to set up an updated, integrated matrix of experimental facilities (i.e. updating the reports NEA/CSNI, 1993, and NEA/CSNI, 1996). Namely, the following facilities are considered in Appendix A3 with a summary table that synthesizes the given tables for:

- ITF – PWR;
- ITF – BWR;
- ITF – VVER;
- Selected SETF (also VVER);
- ITF advanced reactors;
- Containment facilities.

Additional information related to selected facilities also can be found in Tables 3-7 to 3-11 in this chapter.

Table 3-6 – Comparison of merits and issues of SETF and ITF.

Items	SETF		ITF	
	Merits	Issues	Merits	Issues
Geometry & Fluid Conditions	• Possibility to minimize scaling distortion by employing (almost) full-scale conditions	• Distortions may exist depending on facility design; local geometry and/or non-prototypical fluid	• Main reactor configuration / key system components relevant for safety and design studies can be represented • Representation of multi-D pheno-mena and their mutual interactions during a system transient, depending on ITF BC design	• Reduced-scale in volume for all ITFs , with further reductions in height, pressure (temperature), power, and/or the number of loops, depending on ITF-scaling approach • Compensating actions may necessary to minimize scaling distortions; ex. orifice in main circulation loop of PWR ITF to control pressure drop, heat losses, etc.
Initial & Boundary Conditions (I&BC)	• Well defined I&BC to characterize local phenomena by simulating interactions at facility boundary with other components • Hardware control of discharge (critical) flow under steady conditions	• Distortion in interacting phenomena at facility boundary due to BC scale effect(s) • Distortion / difference in evolution of phenomena at location of measurement depending on facility geometry and/or I&BC • Condition range may not fully cover expected response of prototype due to limitation in I&BC, depending on facility design constraints	• Well- defined IC & BC at system level: pressure and temperature distribution, FW and steam flow rate. • Hardware control of break discharge (critical) flow under accident simulation conditions	• Precise definition difficult for local parameters, such as flow rate in each of PWR SG U-tubes and core sub-channels.
Phenomena Representation	• Reproduction or detailed simulation of local (single) phenomenon and specific parameter(s) by avoiding or reflecting influences from the periphery • Characterization of TH behaviour of target component for specific parameter(s) by decreasing or reflecting influences from the periphery • Suitable for observing	• Most SETFs are designed for steady experiments that may not include transient phenomena, while depending on SETF design • Sometimes, SETF design does not fully consider scaling to reference reactor	• Simulation of accident phenomena, in multi-dimensional in general, with mutual interactions through the system transient within the limitations of the ITF design limitations • Component simulation under either of steady operation and	• Distortion in flow rate, energy, and mass-distribution and/or phase separation / stagnation due to constraints in facility design • Time-scale and/or phenomena evolution (ex. mixture level in core) may change in some reduced height ITFs, basically following scaling approach and the

Items	SETF		ITF	
	Merits	**Issues**	**Merits**	**Issues**
	strongly scale-dependent phenomena. including 3-D effect • While most of the SETF is aimed for steady experiments, transient phenomena could be observed: ex. UPTF, CCTF, SCTF for 2D-3-D programme designed to reproduce local prototypical phenomena under both steady- and transient- conditions		transient conditions	method used for facility design
Counterpart Test (Section 3.3)	• Advantages in investigating single phenomenon with various views from complementary facility size and test conditions	• Difficulty in developing counterpart tests • Distortions in I & BC and facility size can affect test results	• Advantages in clarifying influences of scaling and local geometry • Possibility of addressing scaling issues via similar tests even when the reference reactor is different	
Data Measurement	• Data for local phenomena understanding, model / correlation development and code validation • Instrumentation dedicated to characterize target phenomena and easier improvements than for ITF • Spatially precise measurement both for steady and transient phenomena •Possibility of CFD-grade (very precise and high spatial & temporal resol-ution) measurements with specific facility design	• Limited parameters within test planning	• Measurement of both of steady- and transient- phenomena at specified (fixed rather small number of) points • Data for local and system-wide phenomena understanding and code validation (incl. CFD codes)	• Spatially coarse measurements for limited parameters and difficulties in changing/adding instrumentation • Difficulties in simultaneous measuring multiple parameters to understand multi-D two-phase flows • Development of instrumentations that withstand high-pressure and temperature

3.2.2 SETF for phenomena in reactor systems

A Separate Effect Test Facility (SETF) is used to characterize, from a thermal-hydraulic point of view, the reactor component behaviour (SETF-Component test) by characterizing the component responses that are typical of the design function, and the local phenomena and processes (SETF-Basics test) to validate closure relations, USNRC, 1988. In one SETF, one phenomenon or several combined ones can be investigated owing on the facility design and capability.

The main characteristics of a SETF:

- Desirable to have minimum scaling distortions:
  - full/almost full-scale, NEA/CSNI, 1989;
  - prototype/almost prototype fluid conditions;
- Instrumentation dedicated to characterize selected phenomena, Wolfert, 2008;
- Well imposed boundary conditions necessary to simulate interactions with the other reactor components, NEA/CSNI, 1989, and Wolfert, 2008, not simulated in the experimental test-facility.

The SETF scaling distortions (or SETF scaling limits), Karwat et al., 1985, mainly are due to the scaling effect of the external boundary conditions, causing a distortion on the interacting phenomena at the facility boundary. However local geometrical distortions, NEA/CSNI, 1996a, and initial deviations in boundary conditions could be present. Distortions could be present if a non-prototypical fluid is used, D'Auria & Galassi, 2010.

In many cases, the data obtained from a SETF can be used to estimate the full-scale prototype. The direct extrapolation to the prototype, however, requires caution considering scaling limits in the SETF. For application of the code, the data could be useful as a basis to assess the component dynamics (SETF-Component test), develop/improve closure equation (SETF-Basics tests), USNRC, 1988, and to assess uncertainty in prediction at full or 'almost full' scale, NEA/CSNI, 1989.

Counterpart tests are important to assessing the effectiveness of scaling criteria, evaluating the effect of scale distortions, assessing scale-up capability of the experimental data, USNRC, 1988, and determining the scale-up and scale-down capabilities of the computer codes, as also discussed in Section 3.3.

It is important to underline that when a phenomenon is strongly scale-dependent, a SETF experimental investigation should be necessary, IAEA, 2005, and NEA/CSNI, 1993. ITF-SETF coupled analyses, Wolfert, 2008, are useful particularly in analysing scaling issues.

For analyses of the main characteristics and some scaling issues/topics of the RCS-SETF, the main references that should be mentioned and, some, used as a basis of the following Sections are

- CSNI Report No. 161, 1989: Thermo-hydraulics Of Emergency Core Cooling In Light Water Reactors: a State of the Art Report (SOAR), NEA/CSNI, 1989;
- NEA/CSNI/R(93)14: Separate Effects Test Matrix for Thermal-Hydraulic Code Validation, NEA/CSNI, 1993;
- NEA/CSNI/R(96)16: Evaluation of The Separate Effects Tests (SET) Validation Matrix, NEA/CSNI, 1996a;
- Compendium of ECCS Research for Realistic LOCA Analysis, Final Report, USNRC, 1988;
- https://www.oecd-nea.org/dbprog/ccvm/indexset.html.

For the analyses of the main characteristics and some scaling issues/topics of the VVER reactor facilities, the main reference that should be mentioned is:

NEA/CSNI/R (2001)4: Validation Matrix for the Assessment of Thermal-Hydraulic Codes For VVER LOCA and Transients, NEA/CSNI, 2001.
It serves to underline that the large amount of the experimental database precludes the possibility of a complete discussion; however, selected scaling issues/topics are considered and briefly analysed based on available information from the previous references and others mentioned in the list of references (this list is broad enough but shall not be considered exhaustive).

3.2.2.1 Scaling Considerations for Separate Effects Tests (SETs)

The adequacy of the SETF bases, namely the adequacy of the obtained data, is analysed in NEA/CSNI, 1989, and discussed further in NEA/CSNI, 1993, and NEA/CSNI, 1996a. Items to judge the adequacy may include such points as relevance to nuclear safety, data requirement for the development and validation of models, range of conditions, and accuracy and consistency of the measurements, these explanations as well as those in raised in Chapters 2 and 4 also can be used to derive the role of SETF for scaling.

It is important to underline that:

- The influence of the facility scale is observed in many SETF tests, NEA/CSNI, 1989;
- Full-scale/almost full-scale tests clarified that the weight of the LOCA phenomena is influenced by 2D-3-D effects, NEA/CSNI, 1989;
- SETF-Full-scale test facilities, such as the UPTF, are necessary to characterize multi-D phenomena, Wolfert, 2008.

Typical examples of problems encountered with SETs are the phenomena during blowdown, mixing, flow stratification and heat transfer in the fuel bundle. More details are given in the report, NEA/CSNI, 1989.

3.2.2.2 SETF characteristics

As highlighted in the previous section, it is desirable that the SETF is characterized by a minimum scaling distortion with full/almost full-scale and prototype/almost prototype fluid conditions. Although a large number of SETF have been designed and operated in the last decades, it is beyond our target to offer a detailed analysis of each of them. Therefore, only a few examples are considered to underline the main features, a few initial/boundary conditions, and counterpart/similar tests (i.e. SETF vs SETF and SETF vs ITF).

An example of SET is the ACHILLES test facility, NEA Databank, 2015c. The main characteristic is a shroud vessel that contains the test section and the downcomer. The fuel bundle is characterized by PWR- prototypical geometry (rod diameter = 9.5 mm; rod pitch: 12.6 mm; heated length = 3.66 m) with a reduced number of fuel rods (n = 69). In this facility, the core heat transfer phenomena during the reflood phase of a LBLOCA in a PWR have been investigated.

Another example is the IVO/LOOP-SEAL test facility, NEA Databank, 2015e, representing a full-size model of CL and a loop-seal (cross-over leg). The two-phase-loop seal (clearing) phenomena during a LOCA in a PWR were investigated.

Related to the SET analyses of SG thermal-hydraulic behaviour, an example is the PATRICIA (GV) test facility, NEA Databank, 2015b. The facility represented the secondary side of the SG type 51 with 9 tubes in full-size, minimized wall effects, and real support plates. Experiments were conducted on the mixture level and entrainment in the vertical components, and also heat transfer in the SG primary side.

Another example of SETF is the CREARE test facilities, e.g. Crowley et al., 1977, which were used to simulate PWR downcomer ECC bypass during a LBLOCA and to analyse the extrapolation of the results to the full-size plant. Detailed discussion is given in Section 4.3.4.1.

The Slab Core Test Facility (SCTF), USNRC, 1993, USNRC, 1993a, and USNRC, 1988, is a full-height, full-radius, one-bundle-width test facility designed to simulate a 3300 MWt Trojan PWR-4L reactor. This facility permits the thermal hydraulic characterization of multidimensional phenomena. The core-heat transfer phenomena and the ECCS performance at the end of the blowdown and in the refill/reflood phase have been investigated. The main characteristics of the facility, compared with the UPTF and CCTF (ITF) are reported in the Table 3-7, USNRC, 1993a, and USNRC, 1988.

The Upper Plenum Test Facility (UPTF), USNRC, 1993, USNRC, 1993a, and USNRC, 1988, is a full-scale facility to simulate the primary system of a KWU-4L 1300 MWe reactor. Imposed boundary conditions (simulators) are used to simulate the core, the RCPs, the SGs, and the PCV. This facility permits the thermal-hydraulic characterization of multidimensional phenomena. The main characteristics of the facility, compared with the SCTF and CCTF (ITF), are reported in Table 3-7.

The ROCOM, Vattenfall and Gidropress test facilities have been used to characterize the coolant mixing in a PWR downcomer. In particular, ROCOM is a four-loop acrylic-glass test facility that simulates the primary side of the German KONVOI-type reactor. The facility is 1:5 linearly scaled and water at room temperature is used under ambient pressure. The Vattenfall (mixing) test facility is a 1:5 scale model of a W-PWR 3-loop with three loops (one active, and two idle). The lower plenum and the lower 2/3 of the downcomer are made of acrylic glass to effectively observe coolant mixing. The EDO “Gidropress” test facility is a 1:5 scale model of the VVER 1000. The facility is made of metal and a loop with a loop seal and RCP simulator is modeled. The other three loops are short circuits, and only their pressure loss is simulated, Kliem et al., 2007, and Rohde et al., 2005.

SETF is designed to isolate and investigate one phenomenon by minimizing as much as possible the influences of other phenomena. Therefore, counterpart tests (CTs) to compare SETF and ITF for system transients with interactions among phenomena should be difficult. Unavoidable distortions in any facilities (either the ITF or SETF) may pose further difficulties on the SETF/ITF CT.

Table 3-7 - Summary of major features of the SCTF, UPTF, CCTF, USNRC, 1988.

Parameter	SCTF	UPTF	CCTF (ITF)
PRIMARY VESSEL			
General			
Height (m)	8.957	13.5	9.44
Inside Diameter (m)	N/A	4.865	1.084
Downcomer Gap (mm)	250	250	61.5
Design Pressure (bar)	7	22	6
Design Temperature (°C)	350	220	300
Core			
No of Heated/Unheated rods	1872/176	0/49408	1824/224
Rod O.D (mm)	10.7	10.7	10.7
Rod Pitch (mm)	14.3	14.3	14.3
Heated Length (m)	3.66	N/A	3.66
Axial Peaking Factor	1.4	N/A	1.49
Downcomer			
Area (m^2)	0.121	3.62	0.198
Height (m) (bottom of LP to CL nozzle):	8.004	9.2	4.849
Lower Plenum			
Volume (m^3)	1.38	23.9	1.38
Structures	Heater Rod Extension	Piping	Extension of heater rods
Upper Plenum			
Volume (m^3)	1.16	43	2.04
Structures	1/2 Scale Simulation	Full Scale	8/15 Scale Simulation
PRIMARY LOOPS			
Piping			
Number of Loops	1	4	4
Hot Leg Flow Area (m^2)	0.0826	0.44	0.019
Cold Leg Flow Area (m^2)	0.0696	0.44	0.019
Steam Generator			
Number	1	4	4
Type	Steam/Water Separator	Steam/Water Separator	U-Tube and Shell
Number of Tubes	---	N/A	158
Tube O.D./I.D. (mm)	---	N/A	25.4/19.6
Tube Length (m)	---	N/A	15.2
Secondary Pressure (bar)	--	N/A	52
Pumps			
Number	1	4	4
Type	Resistance Simulator	Simulated by Resistance	Resistance Simulator
Break			
Location	Cold Leg	Cold/Hot Leg	Cold Leg, 2 m from PV
Type	100% Offset	Variable up to 100% Offset	100% Offset
Pressurizer			
Number	None	None	None

A CT between two SETFs may have advantages in investigating a single phenomenon with various views from complementary facility-size and test conditions. Examples may include tests with CREARE facilities, Crowley et al., 1977, and Glaeser & Karwat, 1993. Section 2.1.3.1 and Section 4.3 also provide examples.

The consideration of CT in SETF, including some difficulties, may constitute an additional argument that a computer code may be a definitive means to relate SETF and ITF, which can be applied to all these situations. This may complement the discussion in Section 2.2.2.

In the NEA Databank web site of CSNI Code Validation Matrix, such SETFs are described as ACHILLES, G2, ERSEC, IVO, MARVIKEN, NEPTUN, PATRICIA, REBEKA, SMD, THETIS and UPTF [https://www.oecd-nea.org/dbprog/ccvm/indexset.html].

3.2.2.3 Example of SETF scaling distortions

As underlined in the previous sections, the SETF scaling distortions (facility scaling limits) mainly are due to the external boundary condition scale effect, possible initial condition deviations, local geometric distortions and discrepancies in fluid properties.
In relation to the boundary condition distortions, as an example, in the UPTF test '10B-RUN081', steam/water flow phenomena in reflood of PWR Cold Leg Break LOCA, the imposed boundary conditions were:

- No ECC injection;
- Steam- and saturated-water were injected into the core simulator;
- The cold-leg break valve was open;
- The hot-leg break valve and all pump simulators were partially open to establish the desired loop-flow resistances.

Detailed analyses of the boundary conditions used in selected UPTF tests are reported in the SETF NEA databank [https://www.oecd-nea.org/dbprog/ccvm/uptf.htm].

It is important to underline that although some experimental data are available up to reactor scale, geometrical distortions still are present. For example, although UPTF is a full-scale test facility, geometrical distortions are present in the lower plenum for the core-simulators penetrating pipes, NEA/CSNI, 1996a.

The IVO-thermal mixing facility, containing a half of the circumference of the reactor downcomer, is made of acrylic plastic, and operated at atmospheric pressure with loop- and high-pressure injection flows from different cold legs in the area of interest to the Pressurized Thermal Shock (PTS), NEA Databank, 2015d.

The CLOTAIRE test facility (ITF) is an example of a full-scale non-prototypical fluid (Freon-114) test facility used for analyses of the SG secondary side, De Kruijf et al., 2003.

3.2.2.4 Phenomena database coupled with ITF- and SETF-data

Even though counterpart tests, CTs, are not always feasible as discussed before, they are desirable for, and important in assessing the effectiveness of scaling criteria, evaluating the effect of scale distortions, assessing the capability for scaling-up of the experimental data, and assessing the suitability for scaling up and scaling down the capabilities of the codes.

As an example, the phenomena of Entrainment/De-Entrainment in the downcomer and the steam/water interactions in the downcomer have been investigated in UPTF and CCTF. While the ITF tests characterize the thermal hydraulic transient behaviour of these phenomena, the SETF investigation characterizes the effect of the steam flow and the downcomer wall superheat on downcomer water level and entrainment. Considering UPTF test No 2, the downcomer level measured in the two facilities is different; this is due to the scaling and configuration of the downcomer. There are other UPTF-SETF tests that have contributed to our understanding of entrainment and level reduction in the downcomer, NEA/CSNI, 1993.

The comparison between the UPTF and CREARE (1/30, 1/15, 2/15 and 1/15 scale) experimental data, related to the downcomer bypass during a PWR LBLOCA, shows that the thermal-hydraulic behaviour is strongly scale dependent, and that the downcomer flow at full-scale is heterogeneous, Wolfert, 2008. In a paper by D'Auria et al., 1995a, the combined use of ITF- and SETF-data is discussed.

Further experimental work, NEA/CSNI, 1996a, was recommended based on the evaluation of the SET validation-matrix. The following is reported:

> '' *... the limitations are mainly due to measurement techniques and instrumentation needs. It is also necessary to mention that, as new designs of reactors become available, new questions related with new phenomena or extended parameters range may generate new needs for experiments. Need of additional experimental data have been identified as follows: Basic phenomena: pressure drops at geometric discontinuities; Critical flow in valves, Phase separation at branches; Quench front propagation/rewet, fuel rods; Parallel channel instabilities (BWR); Boron mixing and transport; Non condensable gas effect (PWR).*''

In the next section, a few examples are given to show that some new experimental facilities, designed in the recent years, have extended the SETF database included in NEA/CSNI, 1993, and NEA/CSNI 1996a.

3.2.2.5 Example of advanced design related SETF

In relation to the advanced reactor designs, natural circulation and passive systems, IAEA, 2005, experimental data have been gathered by the operation of SETF. Examples are the following:

- NOKO and TOPFLOW (Germany);
- PANTHERS and PERSEO (Italy);
- CLOTAIRE (France) (ITF).

The VVER-1000/V-392, components have been tested using the following SETFs:

- Gidropress SPOT, and
- HA-2 facilities in Obninsk, Russia, IAEA, 2005.

In relation to the ESBWR design, the integral-test programs were conducted in the GIST, GIRAFFE, PANDA, and PUMA facilities. Furthermore, tests were performed in PANTHERS, a full-scale component-SETF of condensers for the ICS (Isolation Condenser System) and in the PCCS (Passive Containment Cooling System) facilities, Gamble et al., 2006, and Woodcock et al., 1999.

The PERSEO is a modification of the PANTHERS IC-PCC. The area of the HX (heat exchanger) pool is the most important dimension for reproducing the expected prototypical physical-behaviour. A 5 m^2 area was used, permitting prototype time-behaviour and good steam/ water separation at the top of pool, Ferri et al., 2005.

The CLOTAIRE test facility (13.6 m high, designed to thermo-hydraulically characterize the SG secondary side, and using Freon-114 as a fluid), will be used to study the operating point of the ESBWR, De Kruijf et al., 2003.

Recently, vendors of SMR (Small Module Reactor) started conducting separate effect tests for their designs. In the United States, the major participants are NuScale, mPower, and Westinghouse. In Korea, there is the SMART design.

NuScale plans to conduct a critical heat-flux test in the Stern Laboratory in Canada. The main purpose is to obtain CHF data so to expand the database for developing the CHF correlation for the operating conditions. Also from this experiment, the temperature rod bundle sub-channel exit can be obtained to determine mixing coefficients. The characteristics of the single-phase and two-phase pressure drop of the assembly for a range of bundle powers and hydraulic conditions also can be obtained. NuScale is planning a test of the helical-coil steam generator using the GEST Facility in SIET (Piacenza, Italy, Colbert, 2013). Their purpose is to obtain large-scale thermal hydraulic data important to designing and operating the NuScale helical-coil steam generator. The information will be used for validating safety analyses and design codes.

mPower is planning component tests of the reactor-coolant pump, the control rod drive mechanisms (CRDMs), fuel mechanical testing, CRDM/fuel integrated tests, the critical heat flux and emergency condenser at B&W Center for Advanced Engineering Research (CAER) in Bedford, Virginia, USA, Arnholt, 2013.

Westinghouse plans to re-use some previous test results from AP600 and AP1000 that apply to the SMR design. They include the ADS (Automatic Depressurization System) tests conducted in Milan, Italy; CMT (Core Makeup Tank) tests and PRHR (Passive Residual Heat Removal) tests conducted in Pittsburgh, USA; PRHR; SPES-2 tests conducted in Piacenza, Italy, and APEX tests conducted in Corvallis, USA. Westinghouse also plans a full-scale Set of the upper plenum and ADS flow paths at the Applied Research Laboratory (ARL) facility at Pennsylvania State University, USA, Wright, 2013.

Regarding the ITF, NuScale plans to conduct the integral system tests (NISTs) using the modified MASLWR facility. The test facility has a 1:3 height and length scale; 1:1 time scale, and a 1:254.7 volume scale with prototypic pressures and temperatures. The facility models the major components, including the reactor vessel, the containment vessel, the reactor building pool, the electrically heated core, the core-shroud with riser, pressurizer, ECCS, the helical-coil steam generator, the decay-heat removal system. mPower built its Integrated Systems Test (IST) facility at CAER in Bedford, Virginia, USA. The test purposes include detailing the heat-transfer phenomena, the performance of the steam generator the LOCA Response, and the pressurizer performance and reactor control. Westinghouse plans to conduct the integral effect tests at the SPES Facility in Piacenza, Italy.

In Korea, various thermal-hydraulic tests had been completed to validate the SMART (System-integrated Modular Advanced Reactor) design, which was generated by KAERI and gained Standard Design Approval (SDA) from the Korean regulatory body in July 2012, Kim et al., 2014. For SETs, the testing of core-flow distribution was performed with the SCOP facility (SMART Core flow distribution and Pressure drop test facility) to verify the core inlet flow rate and pressure distributions of the SMART during 2009 through 2011. An ECC bypass test also was carried out with the SWAT facility (SMART ECC Water Asymmetric Two-phase Choking Test Facility). A Freon CHF test was performed with the FTHEL (Freon Thermal Hydraulic Experimental Loop) facility to construct a database from 5 x 5 rod bundle Freon CHF tests, and to verify the DNBR model in the safety analyses and the core-design codes. The IETs also were performed during 2009 through 2011 by using the small-scale VISTA-ITL experimental facility. This facility has a volume scale ratio of 1/1310 for SMART, and was used to simulate the SBLOCA scenario for the SMART design. In addition, performance-related tests for the natural circulation of the primary system and PRHRS (Passive Residual Heat Removal System) were completed. The test results for SETs (SCOP, SWAT, FTHEL), and IET (VISTA-ITL) were used efficiently during the SMART standard design and licensing, Yi et al., 2013, and Park et al., 2014. At the end of 2012, a new large-scale integral-effect test facility, FESTA (or SMART-ITL), which has a full-height and volume scale ratio of 1/49, finished its commissioning tests. Major thermal-hydraulic tests of the SBLOCA and natural circulation were completed during 2013, Park et al., 2013. Recently, to satisfy domestic- and international-needs for improving nuclear safety after the Fukushima accident, an effort to improve safety was performed and a Passive Safety System (PSS) for SMART was conceptually designed.

A test programme to validate the performance of SMART PSS was launched with an additional test facility to scale-down the SMART PSS, which will be added to the existing SMART-ITL facility.

3.2.3 ITF for phenomena in reactor systems

An Integral Test Facility (ITF) is scaled-down from a specific reference reactor to investigate, USNRC, 1998:

1. The overall system behaviours and the related phenomena and processes;
2. The interaction of two or more components, and
3. The local phenomena that are typical of the overall system design target function.

The ITF can then be defined as a test facility composed, at least, of a heat source and a heat sink connected in a closed circuit by a hydraulic path. Several systems can be connected at this closed circuit, Karwat, 1986.

Considering the scaling approaches used to design an ITF, several distortions will be present causing the partial- or the total failure of properly simulating phenomena – i.e. scaling limits in each ITF, Karwat, et al., 1985. Such distortions are due to scaling methods that may include volumetric scaling, linear scaling (length- and, height-scaling), power scaling, core- operation- mode scaling, system pressure scaling, fluid property scaling, geometrical configuration scaling (such as lumped loop or not). However, for each ITF, certain scaling methods are selected to use during its design.

In general, the data obtained from an ITF cannot be applied directly to full-scale prototype, if the validation of computational tools (code) is not being used. The direct extrapolation at the full-scale prototype requires caution in considering the facility scaling limits and the appropriate methodology; using codes are necessary to do the extrapolation. Examples are reported by Koizumi et al., 1987, and Bovalini et al., 1993.

By applying the code, the data could be used for assessing the system dynamics and predicting the components' interactions, USNRC, 1988.

Counterpart tests are important to assess the effectiveness of scaling criteria, to evaluate the influences of scale distortions, and to assess the scale-up- and scale-down-capabilities of codes within a certain range of validation data, USNRC, 1988, Koizumi et al., 1987, and Bovalini et al., 1993.

To analyse the main characteristics and some scaling issue/topics related to the RCS-ITF, several references were considered as a source of important information on the ITFs, e.g. Karwat, 1985, Karwat & Austregesilho, 1985, Karwat, 1986, USNRC, 1988 (namely, see Appendix A), NEA/CSNI, 1987, NEA/CSNI, 1989, NEA/CSNI, 1996d, NEA/CSNI, 1996c, NEA/CSNI 2000, NEA/CSNI, 2001, including information from the NEA Databank, NEA Databank, 2015.

Table A3-1 in Appendix 3 shows the list of Country, Organization, and ITF-facilities used for analysing phenomena in the reactor coolant system in current designs and advanced ones.

3.2.3.1 Current PWR-related facilities scaling considerations

A generic PWR design is characterized by the primary system, with an open-channel fuel bundle in the reactor core, where the power is generated and transferred to the secondary system through the Steam Generator (vertical U-tube SG, horizontal SG, and the OTSG: viz., the Once-Through SG). Both active and passive systems are connected to the loop piping. An experimental test facility has to reproduce the main components of its layout and the peculiarities of its main design. For example, the B&W design is characterized by OTSG and by a 2×4 Loop configuration. Most of these main features should be reproduced in the test facilities.

Once the scaling approach is defined, a scaling method is used to design the ITFs. For example, the power-to-volume scaling method was used to design such facilities as LOFT, SEMISCALE Mod-1 to 3, LOBI, PKL, LSTF (ROSA–IV), CCTF, BETHSY, SPES, OTIS (a modification of GERDA, Tietsch, 1999), MIST, NEA/CSNI, 1989. Another example is the UMCP (University of Maryland at College Park) 2 x 4 loop that simulates a B & W reactor (USNRC, 1988, namely Appendix A). In this facility, the time scaling is preserved but not the height scaling (core height 1: 3), NEA/CSNI, 1996d. In the SRI-2 facility at Stanford Research Institute, Sursock & Kiang, 1985, the Ishii and Kataoka scaling rationale was applied whilst considering the non-prototypical pressure, e.g. USNRC, 1988 (Appendix A) and Ishii & Kataoka, 1983. The APEX-CE facility, a modification of the APEX facility, was designed using the H2TS methods, Reyes, 2001.

Table A3-2 in Appendix A3 summarizes the scaling information for the main facility for current PWR ITF-related facilities.

Time Scaling

An “Expected time of event 1” (time-preserving approach) is obtained when the distribution of mass and energy along the loop is preserved. This is accomplished via a proportional reduction of power and facility volume, NEA/CSNI, 1989. When the body forces due to gravity are small compared to local pressure differentials an “Expected time of event ≠1” (time not preserved approach) could be used. An example of this application is the simulation of pressure-wave phenomena. Linear-scaling reduction of any dimension of the facility determines time scaling of the facility. However, when the “expected time of event is ≠1”, severe distortion in the heat-transfer process should be expected, NEA/CSNI, 1989.

Table A3-2 in Appendix A3 shows the expected time of event of the facilities used for analysing the designs of current PWRs. For example Semiscale, LOBI, SPES, PKL-III, BETHSY, LOFT, LSTF, wherein a PWR counterpart test was performed, NEA/CSNI, 1996d, are characterized by “expected time of event = 1”, that is time preservation.

In a general study of the similarity criteria and feasibility of LOFT as a PWR natural-circulation simulator, Ishii & Kataoka, 1982, and Ishii & Kataoka, 1984, it was concluded that simulating real time is possible only in single-phase flow by installing an orifice and a horizontal length of pipe that should be of the correct dimension after considering both of the similarity criteria for single-phase and two-phase flows. A general modification of LOFT was recommended, Reeder, 1978, by installing orifices in the hot leg.

Linear Scaling

Since each reduction of the linear dimension of the facility determines a time reduction for the facility (the expected time of the event is different from 1), it is important to consider the length reduction (for example, the loop length) and the height reduction of the facility compared with the prototype. The latter parameter influences the capability of the facility to simulate the gravitational effect that determines natural circulation characteristics through primary loop.

Height Scaling

In general, the reduction of the facility height (linear scaling) determines the extent of reduction in time of the simulated phenomena. Time preservation should be obtained by installing orifices in the loop to control the local and/or system-wide velocity of flow, as also discussed in above “Time Scaling”. For example, in the test facilities LOFT, Ishii & Kataoka, 1984, and APEX-CE, Reyes, 2001, the height is not preserved, but the time scaling is preserved (expected time of event equal 1).

More detailed information is reported in “Scaling Distortion” sub-Sections 2.2 and 3.2.6.

The scaling of the facility height influences the capability for simulating the gravitational effect in the facility; this capability determines the regimes of natural circulation flow. As it is reported in the Table A3-2 in Appendix A3, in general, the height is preserved in those facilities characterized by having the current PWR designs, NEA/CSNI, 1989. In the test facilities LOFT, UMCP, NEA/CSNI, 1989, SRI-2, USNRC, 1988 (Appendix A) and APEX-CE, Reyes, 2001, height is not preserved (LOFT 1:2; UMCP 1:3; SRI-2 1:4; APEX-CE 1:3.45). In the LOFT facility (initially designed to simulate LBLOCA, but used also for SBLOCA), for example, the reduced height of the core and the SG distort the SBLOCA simulation, NEA/CSNI, 1989.

The relative height distribution of the facility components (height distribution ratio), coupled with the relative volume distribution influences the energy and mass distribution along the loop. Of particular interest is the volume versus height plot that is used for qualifying a code nodalization.

Volumetric Scaling

For current PWR reactors, the facility volumetric scaling ranges from 1 : 21.4 (CCTF test facility) to 1 : 1705 (SEMISCALE Mod2 and Mod3). For the B & W design, for example, UMCP is characterized by 1 : 500 volumetric scaling and OTIS is characterized by 1:1686 volumetric scaling, NEA/CSNI, 1989. For the CE-PWR, about 1 : 276 volumetric scaling is considered in the APEX-CE facility, Reyes, 2001.

Considering the influence of the height of the experimental test facilities for correctly simulating natural circulation phenomena, a decrease in the cross-sections of the vertical flow section in general is used for fulfilling the volumetric scaling target, Karwat, 1985.

The volumetric scaling of the facility determines the ratio of wall surface/hydraulic volume. In particular, a decrease of the facility volume, compared with the prototype reactor, causes an increase in the ratio of the wall surface/hydraulic volume. This last parameter affects the structural stored heat and the heat losses. The key role of structural heat in a short term accident, as a LBLOCA, and the key role of the heat losses in a long term accident as SBLOCA are emphasized, NEA/CSNI, 1996d.

Various test facilities are designed to compensate, at least partially, for these effects by using the thermal insulation of the circuit, by using the electrically heated coils surrounding the circuits, and also using the modified decay heat curve, NEA/CSNI, 1996d. For example, heaters and insulation have often been used in many ITFs, such as OTIS, MIST, BETHSY, LSTF and SEMISCALE, NEA/CSNI, 1989. For example, in BETHSY, heat losses are controlled by external heater (heat tracing) system, NEA/CSNI, 2000. In SEMISCALE Mod-3, the core power was increased for some experiments, while SEMISCALE Mod-2A employed external heaters connected to the loop piping. In OTIS, the loop piping passive insulation and guard heaters are used: namely, guard heaters are installed in the HL, surge-line, PRZ and RPV upper head, USNRC, 1988 (Appendix A).

It is emphasized that in an ITF the distribution of the volume (volumetric distribution ratio), in comparison with the prototype distribution, should be preserved. This assures that the facility component volume is proportional to the respective volumes of the prototype. This arrangement, coupled with the preservation of the relative height and the location of the single reactor components (i.e. the core relative height, the height of cold leg and hot leg, and the like), maintains the distribution of energy and mass along the loop.

Fluid scaling

Fluid considered for the simulation of the PWR RCS-facility is, in general prototypical, i.e. water for both the liquid- and steam-phases. For example, all the main RCS-ITFs, used to simulate current- and advanced-PWR, use water as a working fluid. An example of a non-prototypical fluid facility is the DESIRE facility, simulating the Dodewaard NC BWR, which uses Freon 12 as working fluid (reported in Appendix 3).

As for experiments that employ a simulant fluid such as Freon, it is important to confirm the applicability of scaling laws that should include influences of the fluid properties in some scaling factors. A simulant fluid is employed for experiments at low pressures for representing the thermal-hydraulic responses expected for water under high pressures. This approach offers such advantages as the saving of resources and simplification in managing experiments, which may include easy visualization of flow structure (such as flow patterns) in complex geometries.

Justification should be needed when simulant fluid is used to reproduce phenomena during a full transient of LWR accidents because the interactions between gas- and liquid-phases, including phase changes, depend on the fluid properties. The utilization of a simulant fluid is favorable as long as it is confirmed that it is suitable even for in-depth investigation of details of an assigned phenomenon.

Pressure Scaling

The operating pressure of the facility influences the physical properties of the fluid. In a PWR, the primary pressure is at about 15 MPa and, as shown from Table A3-2 of Appendix A3, the primary- system pressure in most cases is preserved. In the SPES facility, however, the primary system pressure is up to 20 MPa. This supports studies of power excursions, NEA/CSNI, 2000. The PKL, CCTF, UMCP, NEA/CSNI, 1989, SRI-2, USNRC, 1988 (Appendix A), APEX-CE, Reyes, 2001, are characterized by reduced pressure. The reasons for employing high/low design pressure are governed by the objectives of experiment, thus following their scaling approach, and not only by budget constraints.

The primary pressure of the CCTF and the PKL test facility, respectively, is 0.6 MPa and 5.0 MPa. The CCTF was designed to simulate the refill and re-flood phases in an LBLOCA, characterized by a low pressure condition. For the PKL test facility, the primary/secondary pressure of about 5.0/6.0 MPa permits analyses of a wide range of temperatures, and also the simulation of relevant phenomena and important phases of many accident transients in the original pressure range below 5 MPa, Umminger et al., 2012. For transient parts developing in HP (high pressure)-conditions, e.g. in the early phase following an SBLOCA, the initial- and boundary-conditions are well controlled and adjusted in the frame of a so-called "conditioning phase" to start the simulated transient under prototype plant conditions at approximately 5.0 MPa. The status of the prototype plant at this pressure level is usually obtained by analyses of the prototype plant.

The secondary pressure-range of the current PWR test facilities is reported in Table A3-2C of Appendix A3.

Nuclear Core Simulator Scaling

The heat-transfer characteristics due to geometry, material and electrical-heating methods and operational mode are important scaling factors to simulate thermal response of nuclear core of reference reactor during reactor accidents and abnormal transients. In fact, the inner structure and materials of simulated fuel rods and their cross-sectional size, such as diameters and pitch (sub-channel size) as well as the heated length and the radial- and axial-distribution of the linear heat rate should determine the characteristics of heat transfer of the core, and thus, the simulation capability of the ITF. The heating methods, such as direct skin (clad) heating and indirect rod-inside heating, also influence the heat-flux transient. Since the electrically heated simulator usually has a much smaller thermal capacity, NEA/CSNI, 1989, than that for nuclear fuels, electrical core power for ITFs should be controlled to compensate for the difference from the reactor in the simulated heat- flux, even during the accident transient.

As reported in Table A3-2B of Appendix A3, the core geometrical data are close to the prototypical value. For example, the CCTF is characterized by an indirect heating method with 2048 electric rods: its diameter and pitch of 10.7 mm and 14.3 mm, respectively, are equal to those of reference reactor. SEMISCALE also employed an indirect heating method with 25 electric rods with a diameter and pitch of 10.7 mm, and 14.3 mm respectively, NEA/CSNI, 1989.

It is emphasized that full core-power scaling is not inevitably selected in the scaling approach. For example, LOFT (nuclear core), SEMISCALE, LOBI, and SPES are designed to simulate the full-power condition NEA/CSNI, 1989. In the SPES test facility, a maximum primary system pressure of 20 MPa supports the study of power excursions; 140 % of the scaled nominal power, NEA/CSNI, 2000. In PKL, LSTF (ROSA-IV), CCTF, BETHSY, OTIS, MIST, UMCP the decay-power condition is simulated, NEA/CSNI, 1989. When the full power condition is not employed, specific operating procedures are used. Examples are the LSTF- and BETHSY-facilities, NEA/CSNI, 1989.

In relation to the axial shape of the core power, for example, in LOBI, Addabbo & Annunziato, 2012, and that the LSTF is chopped cosine while in PKL-III, Umminger et al., 2012, and SPES it is uniform, as shown in Table A3-2B. The LSTF, ROSA-V Group, 2003, and CCTF, USNRC, 1988 (Appendix A), are characterized by 1.49 axial-peaking factor. It is emphasized that LOFT is the only test facility with a nuclear core, NEA/CSNI, 1989.

SG Simulator Scaling

The SG simulator heat-transfer characteristics (secondary-side heat-deposition rate) for the SBLOCA transient are important scaling factors. The SG geometrical data, viz., the diameter and pitch of the inner and outer tubes determine the characteristics of the SG heat transfer and are, in general, close to the prototype values (Table A3-2C in Appendix A3). For example, the BETSHY test facility has 34 U-tubes in each of 3 SGs; the inside/outside diameter and pitch are 19.7/22 mm and 32.5 mm, respectively. They are equal to those of the reference reactor. UMCP is characterized by having two once-through SGs with 28 tubes; their inside/outside diameter and pitch are 30.0/ 31.7 mm and 50.8 mm, respectively. APEX-CE has 2 SGs with 133 U Tubes, the inside/outside diameter of which is 15.42/17.45 mm.

The energy released into the SG secondary side (the heat-deposition rate) during an SBLOCA transient comes just from the simulated core wherein the rate of power generation is pre-programmed or manually controlled, NEA/CSNI, 1989. Since the SG heat removal rate is the primary factor in controlling the primary pressure, the correct simulation of the chronological change in the rate of SG removal is the key factor for the ITFs to simulate the SBLOCA scenario in a chronologically correct manner.

The position of SG relative to core is a very important factor for simulating the natural circulation. For the B&W design, in this regard, it is possible to distinguish differences in the system response between “lowered loop” plant and “raised loop” plant, NEA/CSNI, 1996d. The latter is characterized by a SG above the RPV inlet/outlet nozzle, showing higher natural circulation driving force than in the lowered loop. For example, OTIS and GERDA, which simulates a B&W with a raised loop should have higher natural circulation flux than in MIST and UMCP that has a lowered loop configuration, NEA/CSNI, 1996d.

Number of loop scaling and main coolant lines scaling

In relation to scaling the loop and main coolant lines, it is important to underline the configuration of the reactor loop. For example, in a simplified noding for safety analysis, typically, a three- and four-loop PWR is characterized such that each loop has a single HL, a single CL, a single U-tube SG, and a single pump. A typical B&W-PWR may have in each loop a single HL, a single once-through SG, two CLs, and two RCPs in each loop. Typical CE-PWR has a single HL, a single U-tube SG, two CLs, and two RCPs. This type of layout should be realized in the test facility to adequately simulation data to validate the computer code predictive capability.

In case of multiple loops, the lumped loop representation may influence the core uncovery phenomena/process during SBLOCA, NEA/CSNI, 1989, and may suppress the effect of asymmetric boundary conditions between the individual loops on the evolution of the accident scenario. SEMISCALE, LOFT, LOBI, and LSTF, for example, are characterized by a lumped-loop approach. The LOFT, for example, simulates a four-loop PWR with two loops; one simulating three intact loops, and one simulating the broken loop, Modro et al., 1985; SEMISCALE and LOBI also have two loops; one represents three intact loops, and the other represents just 1 broken loop, NEA/CSNI, 1989. In the LSTF, the four primary-side loops of the reference W-type PWR are represented by two loops of equal volume. SPES, PKL-III, and BETHSY, on the other hand, preserve the number of loops. SPES is characterized by three single loops to simulate W-PWR-3L, while BETHSY is characterized by three single loops to simulate F-PWR-3L, NEA/CSNI, 1989; PKL-III is characterized by four single loops to simulate KWU-PWR-4L, Umminger et al., 2012.

Table A3-2A in Appendix A3 shows the data for the main PWR facility loops .When multiple loops in the ITF represent different number of reference reactor loops, the diameter of loop piping would be different. Since the level of liquid in the horizontal pipe, hot legs, and cold legs influences the condition of the break flow upstream, one should decide how to arrange the elevations of each pipe; center, top, or bottom of the pipes. In any case, a different response between two loops is more or less inevitable when different size loops are employed, as in LOBI.

In relation to the configuration of the prototype loops (for example, W-PWR: 1HL-1CL. or B&W: 1HL-2CL), SPES, for example, simulates a W-PWR design that is characterized by having a single HL, a single CL, a single U-tube SG, and a single pump for the loop; MIST simulating a B&W design is characterized by a single HL, a single once-through SG, two CLs, and two RCPs for loop; APEX-CE simulating a CE-PWR is characterized by a single HL, a single U-tube SG, two CLs, and two RCPs for the loop.

It is interesting to note the addition of 4 CL loop seals in the APEX facility for the new configuration, APEX-CE, Reyes, 2001.

In general, the cross- section of the primary loop piping is reduced due to the volumetric scaling approach, resulting in a great pressure drop when the fluid velocity is preserved. Therefore, an artificial increase in the scaled cross-section (no preservation of the fluid velocity) is made, coupled with the reduction in the pipe length so to best conserve the volumetric-scaling ratio, Karwat et al., 1985. In the LOBI facility, for example, the single-phase steady-state fluid velocity is reduced by a factor 2 compared with the prototype, USNRC, 1988 (Appendix A).

The reduction of the cross Section of pipe may have some influence on the phase separation phenomena/ process in an SBLOCA transient. The transport characteristics in the horizontal pipe are achieved in SPES, BETHSTY, and the LSTF facility by conserving the Froude number, see e.g. NEA/CSNI, 1989, and Zuber, 1980.

The cross Section of the annulus (downcomer width of the scaled RPV of ITF) could be larger than that of the rigorously scaled-down value NEA/CSNI, 1989. For example in the LOBI test- facility, the correctly scaled width of the downcomer is 7 mm, but, to avoid the CCFL problem due to the facility volumetric scaling (1:712), two larger widths are used: 50 mm and 12 mm, USNRC, 1988 (Appendix A). Also, in the CCTF test facility, to avoid the hot-wall effect and since the area of the core bypass area is included in downcomer area, the width of the RPV downcomer is larger (61.5 mm wide) than the correct scaled-down value, USNRC, 1988 (Appendix A).

The pump scaled behaviour in the two-phase condition is important for simulating the SBLOCA facility transient. It serves to underline that scaled-down pump head and torque characteristics usually are different from those of the prototype. Therefore, the analysis of influences on the different response is required. Separate-effect studies are important with regard to this issue NEA/CSNI, 1989, NEA/CSNI, 2000, and Karwat et al., 1985. In the LOBI-MOD2 test facility, for example, two pump simulators are installed at the pump discharge so to simulate the locked rotor resistance of the main-coolant pump. Each pump simulator consists of 2-way ball valves, Ohlmer et al., 1985. The simulation of the pump coast-down could be inertial (for example, LOFT and SEMISCALE), programmed (for example, LOBI, SPES, and MIST) or controlled (for example, LSTF and BETHSY), NEA/CSNI, 1989.

Table A3-2A in Appendix A3 details the main PWR facility primary pump system and loop data. Another important parameter is the depth of the pump loop seal that influences the core water level transient during a loop-seal clearing that typically happens in cold-leg break LOCA, Karwat, 1985. Table A3-2A (in Appendix A3) also reports the depth of the loop seal for different RCS-PWR-ITFs. Detailed results of an analysis of the lessons learned from RCS-PWR-ITF are reported in NEA/CSNI, 1989.

Table 3-8 reports some selected K factors for the facility used to characterize current PWR. The K factor is, in general, defined, D’Auria & Galassi, 2010, as follows:

$$K_{\varphi} = \frac{\varphi \; quantity\; value\; in\; the\; experimental\; test\; facility}{\varphi \;\; quantity\; value\; in\; the\; reactor\; prototype}$$

Table 3-8 - Main PWR facility scaling factor summary.

FACILITY	KQ	Kv	Kh	KP	Kq'	Kn	Kl	KM	KNLO	Kq/KV	Expected Time of event
LOFT	0.013	0.02	0.50	0.969		0.025			0.5	0.6	1
SEMISCALE	0.0005	0.0006	1.00	0.938		0.0005			0.5	0.9	1
LOBI	0.001	0.002	1.00	0.969		0.001			0.5	0.6	1
PKL III	0.0007	0.0094	1.00	0.31		0.006			1	0.07	1
LSTF/ (ROSA IV)	0.003	0.02	1.00	1.0		0.023			0.5	0.1	1
CCTF		0.05	1.00	0.038		0.04			1		1
BETHSY	0.0008	0.008	1.00	1.075		0.008			0.75	0.1	1
SPES	0.002	0.002	1.00	1.25		0.002			0.75	1.0	1
OTIS	0.00005	-	1.00						0.25		1
MIST	0.00009	0.002	1.00	0.969		0.0009			0.5	0.06	1
UMCP	0.00005	0.003	0.33	0.125		0.0003			0.5	0.02	1
GERDA	0.00005	-	1.00						0.25		1
SRI-2	0.00002	-	0.25	0.044		0.0004			0.5		
APEX-CE	0.0002	0.003	0.29	0.173		0.0009			1	0.05	1
PWR	1.0	1.0	1.0	1.0		1.0			1.0	1.0	1

Note 1: the nomenclature for parameters considered in Table 3-8 is the following:

Q (KQ): *Total power*

V (Kv): *Overall fluid volume*

H (Kh): *Height*

P (KP): *Pressure*

q' (Kq'): *Linear Power (average & maximum)*

n (Kn): *Number of fuel rods (electrically heated, except for LOFT)*

l(Kl): *Length of horizontal flow paths (including pipes)*

M: (KM) *Mass inventory*

NLO (KNLO): *Number of Loops*

Note 2: empty columns testify of important scaling parameters not derived at the time of writing of the present document

3.2.3.2 Current BWR related facilities scaling considerations

The BWR design is characterized by a core composed of fuel bundles, each enclosed in a zircaloy channel box, along with a system for circulation of the coolant, jet pumps driven by two external loops (BWR/3 to 6), or internal pumps (ABWR). Steam is generated in the core and an active ECCS, or a passive system, such as an IC (isolation condenser) is installed, Nelson, 2008. Throughout the history of the evolution of the BWR design the layout of components inside the Reactor Pressure Vessel (RPV) has not been changed much, concerning their elevation. Since the core structure is unique the test facility design mainly should consider how to simulate the thermal-hydraulic conditions in the core and thus a fuel bundle in channel box. The main design choice would then be to have a full-sized fuel bundle. Then, the volumetric distribution and height scaling are determined, Karwat, 1985. In many cases, the power-to-volume scaling method was used for the design of ITFs such as TLTA, FIST, ROSA-III, TBL, FIX-II, PIPER-ONE, NEA/CSNI, 1989.

Table A3-3 in Appendix 3 reports the main information on facility scaling for BWR ITF.

Time Scaling

In general, the time-preserving approach is used. Then, ‘expected time of events = 1’, for such ITFs as TLTA, FIST, ROSA-III, and PIPER-ONE, where the counterpart test has been performed, NEA/CSNI, 1996d. We note here that the facility of the ROSA-III was half-height, but the test results were considered “time preserved” because it followed the power-to-volume scaling method. The experimentally observed distribution of the void fraction in the core (the channel box) may have encompassed something different from the prototype, though the chronology of accident scenario (time sequence of events) followed expected one in the reference reactor (BWR/6).

Volumetric Scaling

The volumetric scaling for the facilities ranges between 1 : 2200 (PIPER-ONE) to 1 : 424 (ROSA-III), NEA/CSNI, 1989. Table A3-3 in Appendix 3 reports the volumetric scaling and the primary system volume (m3) for the BWR test facilities.

Similar to the PWR-related ITFs, volumetric scaling determines the ratio of wall surface to the hydraulic volume. The consequent distortion of the release of stored energy from the heat structure determines the extent of steam generation that may cause some distortion in the behaviour of the depressurization rate and the water (mixture) level. The increase in the simulated flow area of the Automatic Depressurization System (ADS) can be used to experimentally compensate for this distortion, Karwat, 1985. Such distortion in the structural stored heat may partially be compensated in FIST and TBL by increasing the ADS flow areas; in PIPER-ONE, external cooling is used to reduce its effect on natural circulation, NEA/CSNI, 1989.

Similar also to the PWRs’ ITFs, it is important to preserve the scaled volume of each single component in comparison with the prototype. Table 3-9 shows the component volumes in the simulated RPV for the FIST, ROSA-III, and PIPER-ONE test-facilities.

Height Scaling

Since the BWR fuel bundle is enclosed in a zircaloy channel box, the choice of the main facility design would be to have a full radial-sized fuel bundle. Then, the volumetric distribution and height scaling would follow it.

Table 3-9 – PIPER-ONE, FIST, ROSA-III, BWR-6 geometric characteristics, Bovalini et al., 1992.

Parameter		**Piper-One**	**FIST**	**ROSA-III**	**BWR-6**
Volume Scaling		**2200**	**620**	**424**	**1**
Height Scaling		**1**	**1**	**0.5**	**1**
Core Fuel Bundle	Array	4 x 4	8 x 8 [a]	8 x 8 [a]	8 x 8 [a]
	Number of bundles	1	1	4	624
	Heated length (m)	3.71	3.81	1.88	3.81
	Maximum Power (MW)	0.25	5.05	4.46	3150
	Local peaking factor	1	1.04	1.1	1.13
	Radial peaking factor	1	1	1.4	1.4
	Axial peaking factor	1.26	1.4	1.4	1.4
Elevation (m)	Top of pressure vessel [b]	13.78	19.42	6.04	21.3
	Main steam line [b]	13.15	15.36	6.04	16.2
	Bottom of active Fuel [b]	2.67	4.49	1.59	5.3
	Top of active Fuel [b]	6.38	8.3	3.47	9.11
	Upper tie plate [c]	4.11	4.11	1.96	4.12
	Break (recirculation pump suction) [c]	-0.88	-1.03	-0.65	-1.07
	LPCS injection [c]	4.37	4.47	2.49	4.47
	LPCI injection [c]	1.59	3.62	2.07	3.6
	ADS and SRV [c]	10.46	10.87	4.45	10.9
Fluid Volume (m^3)	Steam Dome	0.072	0.218	0.317	134.2
	Downcomer	0.042	0.17	0.394	108.4
	Jet Pump / recirculation loops	0.0025	0.024 [d]	0.172	9.3 [d]
	Steam separator	0.01	0.047	0.031	29.3
	Upper Plenum	0.007	0.044	0.124	27.5
	Bundles	0.0145	0.043	0.096	26.8
	Bypass	0.005	0.037	0.06	22.7
	Lower Plenum	0.035	0.088	0.167	54.6
	Guide tube	0.005	0.042	0.057	27
	Total (m^3)	**0.199**	**0.712**	**1.418**	**439.8**
Structure (Area, m^2)	Steam Dome	4.3	5.1	5.8	51.8
	Downcomer	3	8.8	11	327.7
	Jet Pump / recirculation loops	0.3	1.2 [d]	11.2	40.8
	Separator	0.7	2.4	2	47.8
	Upper Plenum	0.3	1	1.4	267.9
	Bypass	0.7	5.1	10.4	686.6
	Lower Plenum	1.1	5.7	13.8 [e]	36.5
	Guide tube	0.3	1.5	3.2	443
	Fuel Bundle	3.9	12.6	30	6325
	Surface Area Total (m^2) [f]	**10.7**	**30.8**	**58.8**	**1902.1**
Structure (Volume, m^3)	Steam Dome	0.185	0.091	0.28	4.84
	Downcomer	0.088	0.129	0.48	41.31
	Jet Pump / recirculation loops	0.058	0.002 [d]	0.19	0.6
	Separator	0.006	0.003	0.004	1.2
	Upper Plenum	0.028	0.019	0.01	3.69
	Bypass	0.014	0.069	0.03	4.25
	Lower Plenum	0.073	0.076	0.39 [e]	5
	Guide tube	0.019	0.002	0.02	4.34
	Fuel Bundle	0.012	0.031	0.086	19.39
	Total (m^3) [f]	**0.385**	**0.391**	**1.053**	**65.23**

***Notes:* a: 62 active rods, b: from bottom of pressure vessel, c: related to bottom of active fuel, d: after loop isolation, e: heater connectors included, f: excluding fuel bundles**

The PIPER-ONE, FIX-II and FIST test facilities are characterized by a full-height bundle; ROSA-III is furnished with four half-height yet radially different full scale fuel bundles to observe the influences of the power level of the fuel bundles; one high-power and three average-power bundles. The consequent reduced rate of steam generation is compensated by halving the flow areas of the inlets' orifices and upper tie-plates, NEA/CSNI, 1989. Table 3-9 gives the relative height-component in the PIPER-ONE, FIST, and ROSA-III test facilities (see also Table A3-3 of Appendix 3).

Fluid scaling

In most cases, the fluid used is prototypical steam-water, as shown in Table A3-3 of Appendix A3. An exception is DESIRE facility that uses Freon-12, De Kruijf, et al., 2003. Freon has a merit in enabling a reduction in the pressure of the facility, but in a complex transient it specific physical properties may affect its evolution, D'Auria & Galassi, 2010. When the fluid is not prototypical, the facility owner should clearly indicate a methodology to extrapolate and to consider limitations in the results at the prototype (see the discussion about fluid scaling in Section 3.2.3.1.)

Pressure Scaling

In general, test facilities operate at a primary pressure close to the prototype pressure of 7.8 MPa, NEA/CSNI, 1989, as shown in Table A3-3 of Appendix A3. Exceptions are the DESIRE facility with a primary pressure range of 8-13 bars because of using Freon 12. The CIRCUS facility has a pressure range of 1-5 bars, and is designed to study the low pressure thermal–hydraulic stability of a natural-circulation BWR, De Kruijf et al., 2002.

Nuclear core simulator scaling

All the test facilities use electrically heated simulation fuels. As for core power, TLTA, FIST, TBL and FIX-II have an installed electrical power suitable for simulating a full power behaviour, NEA/CSNI, 1989, while ROSA-III simulates up to 44% power and PIPER-ONE simulates up to 20 % power – 0.28 MW – NEA/CSNI, 2000.

It is noted that the ROSA-III facility had four full fuel-bundles (284 fuel rods) but of half-height (1.88 m), to study the influences of the distribution of radial power in the core. The axial power profile was a chopped-cosine one.

The core geometric characteristics, as diameters and pitch and heated length, are in general close to the prototypical value, as shown in Table A3-3A in Appendix A3.

Recirculation and jet-pump scaling

It is emphasized that a reduced number of jet pumps are, in general, used in the experimental test facilities. In fact, while in a generic BWR there are around 24 jet pumps, fewer are considered in the experimental test facilities.

For example, as shown in Table A3-3 of Appendix A3, TLTA, FIST and TBL consider only two jet pumps, ROSA III considers four jet pumps, while PIPER-ONE considers only one jet pump simulator working only under condition of natural circulation, NEA/CSNI, 1989.

It also noted that, in general, a few scaled-down jet pumps are inserted in the ITF to observe their influence upon the evolution of the core liquid level during a simulated LOCA, especially a Large Break and Intermediate LOCA when the downcomer liquid level is lower than the jet pump inlet nozzle.

In TLTA, the jet pumps are linearly scaled to height and diameter so to correctly simulate the performance of the recirculation flow coast-down during the early phase of a blowdown. The consequence of having short jet pump (i.e. shorter than the prototype) is a lower hydrostatic head and reduced height that influences the core reflood. In FIST, that is an upgrade of TLTA, these are modified to full height, USNRC, 1988 (Appendix A).

In relation to the external recirculation loops, external driving-area scaled-jet pumps are employed on FIST, ROSA III, and TBL to simulate the entire transient of LOCAs from its onset at the rupture of the recirculation pipe rupture. No external loops are considered in PIPER-ONE because it is designed to investigate natural-circulation in the core and downcomer during the SB LOCA, NEA/CSNI, 1989.

In TBL test facility, the jet pumps are mounted inside a scaled annulus (downcomer), NEA/CSNI, 1989) In ROSA-III, in considering the narrow downcomer gap, the jet pumps are installed outside the pressure vessel, Kumamaru & Tasaka, 1985. Since FIX-II simulates a reference reactor without internal jet-pumps, it has only two external pumps, NEA/CSNI, 1989.

It is noted that the scaling problems should be related to the proper simulation of the behaviour of the centrifugal pump, in a two-phase condition, in a recirculation loop and the performance of scaled-down jet pump(s). We also note that critical flow may happen at jet pump driving the nozzle under conditions of reverse flow and at the pump suction pipe when communicative break, say 200%, has happened in one of two external loops, while flow gradually decreases in the other loop. Coolant flashing due to primary depressurization affects all such phenomena. The simulation of the pump characteristics then is more important in a BWR than in a PWR because the M/G set is designed to attain a gradual coast-down in flow to avoid DNB in the core at an early stage of a transient. Separate effect test facilities (SETFs) may help to address this issue, Karwat, 1985.

Detailed analysis of the results of the lessons learned from RCS-BWR-ITF are reported in the CSNI Report No. 161, SOAR on ECC phenomenology, NEA/CSNI,1989.

Table 3-10 summarizes the scaling factors in the main BWR facility.

Table 3-10 – Summary of the Main BWR facility scaling factors.

FACILITY	KQ	Kv	Kh	KP	Kq'	Kn	Kl	KM	KNJP	Kq/KV	Expected Time of event
TLTA	0.002	0.002	1	0.949		0.001				1.228	1
FIST	0.002	0.001	1	0.949		0.001				1.705	1
ROSA-III	0.001	0.002	0.5	0.923		0.005				0.506	1
TBL	0.003	0.003	1	0.923		0.002				1.02	1
FIX-II	0.0009	0.0007	1	0.949		0.0007				1.24	1
Piper-1	0.00007	0.0003	1	0.949		0.0003				0.215	1
DESIRE	0.00001		0.5	0.167		0.0007					
CIRCUS	0.000003		1	0.064		0.00008					
BWR	1.0	1.0	1	1.0		1.0				1.0	1

Note 1: the nomenclature for parameters considered is provided in Table 3-8, in addition:

NJP (KNJP): Number of Jet pumps

Note2: empty columns testify of important scaling parameters not derived at the time of writing of the present document

3.2.3.3 Current PWR and BWR facility used for ISP

The International Standard Problem (ISP) of the NEA/CSNI is a long-term international cooperation activity wherein scaling is one of the key topics of study, see NEA/CSNI, 2000, with its list of selected ISP reports. Typically, within each of these ISP reports, a summary of the scaling-related discussions and findings is given; a full introduction of them is beyond the scope of the present S-SOAR (although an attempt is made to take into account those scaling findings). The following are snap-shots of the information:

– Table 3-11 deals with features of ISP in the area of RCS thermal-hydraulics, taken primarily from NEA/CSNI, 2000, and also from those for ISP-50 NEA/CSNI-ISP-50, 2012.
– Table 3-12 shows the scaling information related to the ATLAS facility used for ISP-50 NEA/CSNI-ISP-50, 2012.

Table 3-11 – DBA related SETF and ITF ISP and scaling information, NEA/CSNI, 2000, and NEA/CSNI-ISP-50, 2012.

No	Type	ISP Title	Facility	Relevant Scaling Consideration
ISP 01	SET	Straight Pipe Depressurization Experiment (Edwards' Pipe)	Edwards' Pipe	Basic separate effects test
ISP 02	INT	Standard Problem 2 (Semiscale Test 1011)	Semiscale	• Volume- to-break area equal in volume to that of PWR • other parameters not scaled; facility should yield data only for comparison to analyses
ISP 03	SET	Comparison of LOCA Analysis Codes	CISE blowdown test rig	Basic separate-effects test
ISP 04	INT	UNITED STATES STANDARD PROBLEM 4 / INTERNATIONAL STANDARD PROBLEM 8 (Simulation of Semiscale MOD 1 Test S-02-6)	Semiscale MOD 1	• Volume- and power-scaling 1/2000 related to PWR • Length of fuel rod simulators equivalent to approximately half of that of commercial PWR fuel -rods • Elevation of steam generators relative to the elevation of the pressure vessel with core simulator preserved
ISP 05	INT	UNITED STATES STANDARD PROBLEM 7 / INTERNATIONAL STANDARD PROBLEM 5 (Non-nuclear Isothermal LOFT Blowdown Test L1-4)	LOFT	• Volume and flow area scaling 1/50, related to PWR • core simulator with flow-restricting orifices representative for nuclear core • Heat-up of the pre-pressurized system by adding energy from running recirculation pumps
ISP 06	SET	Determination of Water Level and Phase Separation Effects During the Initial Blowdown Phase	Blowdown test facility at the Battelle Institute - Frankfurt, Germany	No specific considerations given to typical separate- effects test
ISP 07	SET	Analysis of a Reflooding Experiment	ERSEC LOOP	• Analytical experiment, two-phase thermal-hydraulic oriented exercise • 0.3 MPa total pressure, assumed as constant
ISP 08	INT	Semiscale MOD 1; Test S-06-03 (LOFT Counterpart Test)	Semiscale	• Volume- and power-scaling 1/2000 related to commercial PWRs • Full elevation-length of fuel-rod simulators • Elevation of steam generators relative to that of the pressure vessel with core simulator preserved • Increased steam generator resistance to compared with LOFT reactor situation

No	Type	ISP Title	Facility	Relevant Scaling Consideration
ISP 09	INT	LOFT Nuclear Experiment L3-1	LOFT	• Volume and flow area scaling 1/50 related to PWR • Length of nuclear core equivalent to 1/2 of commercial PWR fuel • Reactor scrammed 2 s before initiation of blowdown process to protect nuclear- fuel elements
ISP 10	INT	Refill and Re-flooding Experiment in a Simulated PWR Primary System (PKL)	PKL	• Volume and power scaling 1/134 related to commercial PWRs • Full elevation length of fuel-rod simulators • Elevation of steam generators relative to the pressure vessel with core preserved • Cross Sections and lengths of loops designed to preserve nominal losses in pressure • break location representative for cold leg break
ISP 11	INT	LOFT Nuclear Experiment L3-6/L8-1	LOFT	• Volume and flow area scaling 1/50 related to commercial PWRs • Length of nuclear core equivalent to 1/2 of commercial PWR fuel • reactor scrammed 5.8 s before initiation of blowdown process to protect nuclear fuel elements
ISP 12	INT	ROSA-III 5% Small Break Test, Run 912	ROSA-III	•Power scaling 1/864 corresponding to BWR-6 type Boiling Water Reactors • Volume scaling 1/437, all nominal flow rates scaled 1/424 • rod bundles half the length of commercial BWR-6 fuel
ISP 13	INT	LOFT Nuclear Experiment L2-5	LOFT	• Volume and flow area scaling 1/50 related to commercial PWRs • Length of nuclear core equivalent to 1/2 of commercial PWR fuel • Reactor scrammed automatically upon initiation of blowdown process
ISP 14	SET	Behaviour of a Fuel Bundle Simulator during a Specified Heat-up and Flooding Period (REBEKA Experiment)- Results of Post-Test Analyses	REBEKA	• Cooling conditions scaled proportional to number of fuel-rod simulators • Dimensions of rods and cladding material identical to PWR
ISP 15	INT	LOCA Experiment in the Swedish FIX-II Facility Related to BWRs	FIX-II	• Power and volume scaling 1/777 corresponding to Swedish BWR Oskarshamn-2 • full length rod bundles
ISP 18	INT	LOBI-MOD 2 Small Break LOCA Experiment A2-81	LOBI	• Volume and flow area scaling 1/712 related to commercial PWRs • Length of fuel rod simulators equivalent to commercial PWR fuel •Relative elevations of components preserved to properly study natural-convection processes
ISP 19	SET	Behaviour of a Fuel Rod Bundle during a Large Break LOCA Transient with a two Peaks Temperature History (Phébus Experiment)	Phébus test facility	Nuclear test, but relevant scaling ratios unclear
ISP 20	INT	Steam Generator Tube Rupture in the Nuclear Power Plant DOEL-2, Belgium	DOEL-2, 2-loop PWR, 1,187MWth (392 MWe), commissioned in 1975	Full-size 2-loop PWR designed by Westinghouse (commercial Nuclear Power Plant)
ISP 21	INT	PIPER-ONE Test PO-SB-7 on Small Break LOCA in a BWR Recirculation line	PIPER-ONE	• Power scaling 1/13,500 related to BWR-6 plant • volume scaling 1/2,200 • full length core simulator • elevations of major components preserved • flow cross Sections corresponding to volume scaling (one-dimensionality of test rig!)

No	Type	ISP Title	Facility	Relevant Scaling Consideration
ISP 22	INT	Loss of Feed-water Transient in Italian PWR (SPES Test SP-FW-02)	SPES	• power and volume scaling 1/427 corresponding to PWR-PUN, Westinghouse 312 type • full length core simulator, power level and nominal flow rates scaled 1/427 • elevations of all components preserved 1/1 to simulate gravitational head
ISP 25	SET	ACHILLES Best-Estimate Natural Reflood Experiment with Nitrogen Injection from Accumulators	ACHILLES test facility	• full length of fuel rod simulator corresponding to commercial PWR fuel rods • elevations of rod bundle, downcomer and upper plenum preserved
ISP 26	INT	ROSA-IV LSTF 5% Cold Leg Small-Break LOCA Experiment	ROSA-IV LSTF	• 1/48 volume scaled main components and loop system • number of fuel rod simulators and decay power scaled 1/48 (however, nominal full power not achievable) • preservation of flow regime transition anticipated by hot leg diameter specification
ISP 27	INT	BETHSY Experiment 9.1B; 2" Cold Leg Break without HPSI and with delayed ultimate procedure	BETHSY	• power and volume scaling 1/100 corresponding to FRAMATOME BWR of the 900 MWe class • full length core simulator, power level and nominal flow rates scaled 1/100 • vertical elevations of all components fully preserved to simulate gravitational head • loop piping diameter of hot legs dimensioned to preserve Froude number criterion of full size plant
ISP 33	INT	PACTEL Natural Circulation Stepwise Coolant Inventory Reduction Experiment	PACTEL	• 1/ 305 volume- and power- scaled model of Russian-designed PWR of the VVER-440 type • maximum power 1 MW equivalent to 22 % of scaled full power • full length fuel rod simulators • elevation of main components including loop seals preserved • diameter (overall height) of steam generators reduced
ISP 38	INT	BETHSY Experiment 6.9c: Loss of Residual Heat Removal System during Mid-Loop Operation	BETHSY	• power and volume scaling 1/100 corresponding to FRAMATOME BWR of the 900 MWe class • full length core simulator, decay power level and nominal flow rates scaled 1/100 • elevations of all components preserved 1/1 to simulate gravitational head • loop piping diameter of hot legs dimensioned to preserve Froude number criterion of full size plant
ISP 43	SET	Rapid Boron Dilution Test	2x4 Thermal-hydraulic Loop Facility (UM 2x4 Loop)	• volume scaling 1/500, reduced component elevation levels • hydraulic diameter ratios of downcomer 1/5 • cold leg internal diameter ratio 1/9 • downcomer nominal velocity ratio 1/18 • scaling of separate effects test based on Reynolds, Strouhal and Schmidt numbers
ISP 50	INT	ATLAS Test, SB-DVI-09: 50% (6-inch) Break of DVI line of the APR1400	ATLAS	The ATLAS is a large-scale thermal-hydraulic integral- effect test facility with a reference plant of APR1400 (Advanced Power Reactor, 1400MWe), which is under construction in Korea. It has a scaling ratio of 1/2 in height, and 1/288 in volume with respect to APR1400; a summary of scaling ratios of the major parameters is shown in Table 3.12. Three-level scaling methodology consisting of integral scaling, boundary-flow scaling, and local- phenomena scaling was applied to the design of ATLAS, Ishii et al. 1998.

Table 3-12 - ISP-50 relevant scaling considerations - Summary of the scaling ratios of the ATLAS, NEA/CSNI-ISP-50, 2012*.

Parameters	Scaling law	ATLAS design
Length	l_{OR}	1/2
Diameter	d_{OR}	1/12
Area	d^2_{OR}	1/144
Volume	$l_{OR}\ d^2_{OR}$	1/288
Velocity	$l^{1/2}_{OR}$	$1/\sqrt{2}$
Time	$l^{1/2}_{OR}$	$1/\sqrt{2}$
Flow rate	$l^{1/2}_{OR}\ d^2_{OR}$	1/203.6
Core ΔT	ΔT_{OR}	1
Core power	$l^{1/2}_{OR}\ d^2_{OR}$	1/203.6
Heat flux	$1/l^{1/2}_{OR}$	$\sqrt{2}$
Power/volume	$1/l^{1/2}_{OR}$	$\sqrt{2}$
Pressure drop	l_{OR}	1/2
Pump head	l_{OR}	1/2
Core rod diameter	1	1
SG U-tube diameter	$l^{1/2}_{OR}$	$1/\sqrt{2}$
No. of core rods	d^2_{OR}	1/144
No. of SG U-tubes	-	1/72

**** The right column indicates the value adopted (in the ATLAS design) for the parameter listed in the central column (e.g. making reference to the first row, length is scaled with the scaling ratio equal to 0.5).***

3.2.3.4 Scaling considerations related to VVER

In relation to the VVER reactors, the main considerations valid for the PWR can be applied. The Power/Volume ratio and Power/Mass ratios are considered in attempting to preserve time in the measured sequence of main events, NEA/CSNI, 2001. The power-to-volume scaling method, coupled with a full-height approach, was used for designing PACTEL, PMK, and PSB, e.g. see D'Auria et al., 2005, and D'Auria & Galassi, 2010, and the ISB test facilities, IAEA, 1995.

Table A3-4 in Appendix A3 shows the scaling information for the main facilities of VVER related ITFs.

In relation to the time scaling, for example, PACTEL, PMK, PSB, D'Auria et al., 2010, ISB, IAEA, 1995, the test facilities are characterized by an 'expected time of event equal to 1'.

In relation to height scaling, all the facilities considered are full-height scales except for the PM-5 with 1:5 scale, NEA/CSNI, 2001.

In general the fluid considered for the facility operation is a prototypical one, NEA/CSNI, 2001.

In relation to the scaling the nuclear-core simulator, considering the reference design (VVER-440, or VVER-1000), all the parameters that characterize the core heat transfer are preserved in general, by using electrically-heated rods, (Table A3-4B in Appendix A3).

In relation to scaling the number of loops, a different approach was taken in the following four ITFs, NEA/CSNI, 2001:

a) The reference plant (VVER 1000) is modeled by two loops in the ISB-VVER facility; one intact and one broken loop. The intact loop includes three SGs; the broken loop includes one SG.
b) In the PMK-2 facility, six loops of the plant (VVER 440) are modeled by one loop.
c) In the PSB, four loops of the plant (VVER-1000) are modeled with four identical loops.
d) In the PACTEL facility, six loops of the plant (VVER 440) are modeled with three separate loops.

Furthermore, in PACTEL facility, volumetric scaling and Froude scaling were used for designing the hot leg and cold leg, Kouhia et al., 2012. In the PMK-2, the volumetric scaling and Froude number were used together with preserving the loop seals elevation, while the length of the prototype is not maintained, Ezsol et al., 2012.

Considering the information available in NEA/CSNI, 2001, Table A3-4A in Appendix A3 shows the data for the primary recirculation pump (or pumps) and for the loop (or the loops).

The important feature of the VVER reactor lies in the horizontal SG: based on the information available in NEA/CSNI, 2001, Table A3-4C in Appendix A3 gives the main characteristics of the heat exchangers (i.e. horizontal SG) of the facilities involved.

Table 3-13 summarizes the main VVER facility scaling-factors.

Table 3-13 - Main VVER facility: Scaling factor summary.

FACILITY	KQ	Kv	Kh	KP	Kq'	Kn	Kl	KM	KNLO	Kq/KV	Expected time of event
PACTEL	0.000333		1	0.510		0.00283			0.75		1
PMK-2	0.000221		1	0.787		0.000374			0.25		1
REWET-III	0.00001		1	0.0223		0.000374			0.25		
KMS	0.01		1	1.15					1.0		
PSB	0.005		1	1.27		0.00330			1.0		1
PM-5	0.000116		0.2	0.0191		0.00138			0.25		
SB	0.000333		1	1.02					0.5		
ISB	0.0006		1	1.59		0.000374			0.5		1
BD				0.0636					0.25		
VVER 1000	1		1	1.0		1.0			1.0		1
VVER 440	0.458		1	0.781		0.773			1.5		1

Note1: symbols are defined in Table 3-8

Note2: empty columns testify of important scaling parameters not derived at the time of writing of the present document

3.2.3.5 Advanced-design-related ITF scaling considerations

Passive and other advanced designs for a water-cooled reactor are characterized by new kinds of phenomena and accident scenarios, and have other features beside their common features with the present-generation reactors, PWR, BWR and VVER.

The new kind of phenomena and accident scenarios given in NEA/CSNI, 1996b, are related to the following:

a) Containment process and interactions with the RCS (reactor coolant system),
b) Low pressure phenomena, and
c) Phenomena related specifically to new system components or reactor configurations.

In relation to the common features with current reactor designs, it is observed that they may have different rankings in the phenomena. Therefore, review of the experimental database available for the current-reactor designs, indicated that further experimental investigations are necessary to characterize the thermal-hydraulic behaviours that are specific to advanced reactors. The phenomena of relevance for advanced design have been investigated extensively, NEA/CSNI, 1996d, IAEA, 2012, IAEA, 2005, IAEA, 2009, and IAEA, 2014. The last document, for example, summarizes detailed analyses done within the IAEA ICSP (International Collaborative Standard Problem) on Integral PWR Design, Natural Circulation Flow Stability, and Thermo-hydraulic coupling of the Primary System and Containment during Accidents.

Several ITFs have been used to investigate natural-circulation phenomena and the thermal-hydraulic response of passive safety systems in advanced reactor designs, IAEA, 2005, IAEA, 2009, IAEA, 2012, and IAEA, 2014.

A list of ITFs used for the analyses of the advanced reactor designs are given in Table A3-5 in Appendix A3.

To analyse the main characteristics and some scaling issue/topics related to the advanced reactor RCS experimental-test facilities, the previous and other references, reported in the reference list, have been considered here.

For designing experimental test facilities for simulating advanced reactors, such scaling methods as H2TS, Zuber, 1991, and three-level scaling, e.g. Ishii & Kataoka, 1984, and J NED Special Issue, 1998 (paper by Ishii), are used. For example, for the design of the APEX, Reyes t al., 1998, and the for the OSU-MASLWR, Reyes et al., 2003, and IAEA, 2005, the H2TS method was used, while for the design of ATLAS, Choi et al., 2014, SNUF, Bae et al., 2008, PUMA, J NED Special Issue, 1998, (paper by Ishii), VISTA-ITL, Park et al., 2014, FESTA, Park et al., 2013, the three-level scaling method was used. A scaling study along with scale distortions for test facilities for AP-600 was performed for USNRC, Wulff & Rohatgi, 1999.

Existing facilities were also used with some modifications for simulating the advanced design. Examples are SPES-2 and ROSA-LSTF for the AP600 simulation. In particular for AP600, Kukita et al., 1996, JAERI implemented the following modifications to the ROSA-LSTF under an agreement with the USNRC:

– Add two CMTs (core makeup tanks);
– Add one PRHR (passive residual heat removal), and one IRWST (in-containment refueling water storage tank);
– Add 4-stage ADS (automatic depressurization system), with catch tanks for the stage-4 valves;
– Add connecting lines for the above components. These included PBL (pressure balance lines), the discharge lines for the CMTs and the IRWST, and DVI (direct vessel injection) lines;
– Replace the existing PRZ (pressurizer) with a full-height and large-volume one, specific to AP-600;

- Add a stand pipe to each of two ACC (accumulator) tanks to allow nitrogen discharge following the discharge of the scaled-water inventory;
- Reduce the depth of the cold-leg loop seals;
- Increase the flow paths between the upper plenum and the upper head, and between the upper head and the downcomer.

The modified LSTF was a 1/30.5 volumetrically scaled full-height model of AP-600, while the existing components were not exactly volumetrically scaled.

The SPES test facility was modified to simulate the AP-600 reactor. The scaling criteria that were applied to the design of the SPES-2 facility are reported by Bacchiani et al., 1995.

Another example is the PACTEL facility that was modified for simulating an EPR-like reactor. Detailed analyses of this facility are reported by Kouhia et al., 2012.

As noted before, the test facility designed to simulate a PWR design has to reproduce the main design peculiarities of the main components, including its configuration. However, the extent of modification should depend on the reactor design and the target phenomena/process. For example, the MASLWR (Multi-application Small Light Water Reactor) is characterized by an integral design wherein the core, HL riser, CL, downcomer, PRZ, and the helical SG are all integrated in the RPV. Otherwise, the AP600 (and AP1000) design is characterized by two loops each one having one U tube SG, one HL and two CL. Then, the modification was possible of existing ITFs, such as the LSTF and the SPES, for simulating the configuration of the AP600 reactor.

It is difficult to accurately modify a given ITF to simulate the reactor layout of AP-600. However, a facility designed to model integral reactors, such as OSU-MASLWR, including their primary circuit and containment, can be used to investigate phenomena dealing with interactions primary/containment or phenomena/processes typical of an advanced design as AP-600. In those cases, the reactor layout and the target phenomena/processes are key considerations for characterizing the facility application to other design different from the reference reactor.

Time Scaling

As shown in Table A3-5 of Appendix A3, ITF are not always characterized by a time-preserving approach (i.e. the expected time of event $\neq 1$). For example, while the SPES-2, ROSA-AP600, OSU-MASLWR are characterized by the time preserving scaling (expected time of event = 1), ATLAS (= 1:$\sqrt{2}$), APEX (= 1:2) and VISTA-ITL (= 1:1.664) are characterized by non-time-preserved scaling.

The lack of time preservation, in general, may introduce an additional distortion, other than unavoidable scaling distortions that already characterize the scaled facility. Related discussion appears in Sections 2.2, 3.2.6, 4.1.4.9, and 4.3.4.3. For nuclear-reactor safety applications, anyhow, the influences of such an additional distortion, although ‘undue’, should be well understood.

Height Scaling

The ROSA-AP600, SPES-2, FESTA, and IST test facilities, are height-preserving scaling facilities with respect to their reference reactor, while ATLAS (1 : 2), SNUF (1 : 6.4), APEX (1 : 4), OSU-MASLWR (1 : 3), VISTA-ITL (1 : 2.77) are reduced-height facilities. It is noted that for OSU-MASLWR, though this is a reduced height facility, the time scaling is preserved. Orifices typically used to achieve time preserving in a reduced height facility, Reyes, 2010.

Volumetric Scaling

Volumetric scaling ranges between 1: 30.5 (ROSA-AP600), and 1: 1310 (VISTA-ITL). The former was test facility has been designed to simulate the AP600 design, while the latter has been designed to simulate the SMART design.

Fluid scaling

In general, the fluid considered for the facility operation is, as in all the concerned prototypes steam and water; see Table A3-5 in Appendix A3.

Pressure scaling

In relation to the pressure scaling, in general the facilities are full-pressure ones. APEX, SNUF and PACTEL-PWR are reduced-pressure ones.

Nuclear core simulator scaling

The nuclear-core simulators, the main characteristic of facilities, are reported in Table A3-5B of Appendix A3.

Number of loops scaling

For the scaling of number of loops, the facility, in general, preserve the number of loops and their configuration as per prototype. For example ATLAS, and SNUF reproduce the one-HL / two-CL loop structure typical of the APR-1400; APEX, SPES-2 reproduces the 1HL-2CL loop structure typical of the AP-600. ROSA-AP-600 has only one CL instead of two as in the AP-600 reference reactor. It is to highlight the integral small-modular reactor and the related configuration of the integral test facility. For example, the OSU-MASLWR test facility reproduces the reference reactor integral configuration; an ascending hot leg riser (HL) and a descending annular downcomer (CL), are integrated in the system. The fluid, after passing through the core, ascends along the HL, and then descends, passing through the SG and the annular DC, arriving at the lower plenum and then the core, Demick et al., 2007.

Table 3-14 summarizes the main advanced PWR facilities scaling factors.

3.2.4 SETF for phenomena in containment

The scope of the present SOAR does not include scaling topics related to severe accidents involving degradation of the core. In addition, the dimensions of the experimental database dealing with all situations expected in the containment may preclude the possibility of in-depth discussion. Therefore, this Section focuses on some highlights on the scaling techniques related to the DBA phenomena in PCVs (Primary Containment Vessels).

Table 3-14 – Scaling Factor Summary for Main advanced PWR Facilities

FACILITY	KQ	Kv	Kh	KP	Kq'	Kn	Kl	KM	KNLO	Kq/KV	Expected Time of event
PWR PACTEL	0.0003		RPV/core: 1:1 SG: 1:4 PRZ: 1:1.6	0.500		0.0028			0.500		
ATLAS	0.0005	0.0016	0.500	1.25		0.0078			0.500	0.336	0.707
SNUF	0.0001	0.0006	0.156	0.0500		0.0051			0.500	0.110	
APEX	0.0002		0.250	0.173		0.00094			0.500		0.500
SPES-2	0.0024		1.00	1.25		0.0019			0.500		1.00
ROSA-AP600	0.0026	0..033	1.00	1.00		0.0198			0.500	0.078	1.00
OSU-MASLWR	0.0002		0.333	0.713		0.0011			-		1.00
VISTA-ITL	0.0002	0.0006	0.361	1.08		0.0007			-	0.3377	0.601
FESTA	0.0008	0.014	1.00	1.13		0.006			-	0.0546	
IST			1.00						-		1.000
EPR	1.12	1.30	1.00	0.969		1.25			1.00	0.860	1.00
APR 1400	1.05	1.30	1.00	0.969		1.12			0.500	0.810	1.00
AP600	0.511	0.683	1.00	0.969		0.75			0.500	0.748	1.00
MASLWR	0.0395		1.00	0.538		0.124			-		1.00
SMART	0.0868		1.00	0.938		0.295			-		1.00
PWR	1.00	1.00	1.00	1.00		1.00			1.00	1.00	1.00

Note1: symbols are defined in Table 3-8

Note2: empty columns testify of important scaling parameters not derived at the time of writing of the present document

Over the past few decades, several facilities have been used to analyse the PCV behaviour in DBAs and beyond-DBAs. To analyse the main characteristics and some scaling subjects related to the SETF and ITF of the PCV under DBA conditions, several documents reported in the reference list were considered, NEA/CSNI, 1986, and NEA/CSNI, 1989a. The following documents are related to the main topic of this Section: NEA/CSNI, 1994, NEA/CSNI, 2009, Karwat, 1985a, Karwat, 1986, Karwat, 1992, IAEA, 1994, and Fischer et al., 2003 (related to SCACEX).

As reported in NEA/CSNI, 1999, several different containment-designs have been developed, specific to each reactor types, PWR, BWR, and VVER, as follows:

- PWR: a) large dry containment, b) ice condenser containments, c) sub-atmospheric containments, and, d) containment for PHWRs (pressurized heavy-water reactors).
- BWR: relatively small volume PCV with pressure-suppression system.
- VVER: a) VVER 440/230, b) VVER 440/213 - Bubble condenser and c) VVER 1000.
- New NPP designs: a) passive Simplified BWRs, b) passive Simplified PWRs, and c) EPR.

Considering the interest of the international community in SMRs (Small Modular Reactors), the containment type and the primary/containment coupling strategy typical of advanced passive SMRs also must be considered, Reyes et al., 2007, as also understood since the early 90, e.g. D'Auria et al., 1992a. In fact, various analyses on the capability of best estimate thermal-hydraulic system-analysis code to simulate natural circulation phenomena and primary/containment system coupling in SMR designs have been completed or are in progress, see e.g. IAEA, 2014, Mascari et al., 2011, Mascari et al., 2012, and Mascari et al., 2012a.

In general, a PCV of a generic large, dry PWR consists of several hydraulic volumes, each of which contains multi components. They are thermally coupled with active/passive heat structures, and are hydraulically connected through specific flow paths, so creating a complex 3-D flow network. Following the terminology used in NEA/CSNI, 1999, three "components" should then be physically characterized in the PCV analysis as:

1) Atmosphere: constituent: gases, two-phase mixture, liquid droplets, and solid particles
2) Water Pool: constituent: liquid, two-phase mixture, gas bubbles, and solid particles
3) Structure: constituent: concrete and steel

A component could be considered as a generalized volume filled with a general composition (gas, liquid, solid), and delimited by a surface, NEA/CSNI, 1999, where energy and mass exchange can take place:

1) Within the volume (source/sink, exchange between the constituents within a volume);
2) On the surface between the volumes;
3) Between the depth of a volume, and a surface or the depth of another volume (emitting/absorbing thermal- and gamma-radiation).

Inside a component several phenomena may act at the same time.

We note that very small solid particles or liquid droplets suspended in gas phase constitute an aerosol (range size from 0.01 μm to 20 μm), NEA/CSNI, 2009.

For thermal-hydraulic characterization of PCV-phenomena, experimental tests are necessary to develop a PCV experimental-database, useful for the analysis of prototype physical-phenomena and for validating the PCV computational tools. Investigation of the phenomena and processes (several phenomena are involved in a process) related to the previous three "components" is necessary to physically characterize the PCV behaviour. The processes occurring in PCV are in general a) transport, b) mixing, c) heat- and mass-transfer, and, d) pressurization/depressurization, NEA/CSNI, 1999.

Following the terminology of NEA/CSNI, 1999, and in agreement with the definition given in the sections of 3.2.2 and 3.2.3, the SETF for PCV (PCV-SETF) is a test facility designed to investigate the following phenomena:

1. reactor-components' behaviour (SETF-Component test) as typical responses due to the design function,
2. local phenomena (SETF-Basics test) to validate closure relations.

One phenomenon (PCV-SET) or several combined phenomena (PCV-COM) can be investigated in one SET. "PCV-COM" is a name used by NEA/CSNI, 2014, to represent facilities where more than one phenomenon is investigated (combined effect).

PCV-ITF then can be defined as a test facility to experimentally investigate all the three components above.

The PCV facilities wherein DBA phenomena are investigated, consist of PCV-ITF tests that show the importance of the heat transfer and compartments interaction, and PCV-SETF tests that have been designed to physically characterize the heat transfer, flow resistance and jet impingement, NEA/CSNI, 1989a. Examples of heat transfer PCV-SETF are the ECOTRA-I and -II experiments. An example of the flow-resistance test is the REBECA experiment. Jet impingement tests were performed in different facilities such as MARVIKEN, a large-scale test facility.

In the recent years, several PCV SETFs and COM facilities have been designed and operated, as reported in NEA/CSNI, 2014. Examples are the TOSQAN and MISTRA that are part of the ISP set of facilities. The TOSQAN, MISTRA, and THAI (PCV-INT) facilities are used for the ISP-47 to investigate gas stratification and its mixing with gas jet. In particular, the TOSQAN facility is employed for the thermal-hydraulic characterization of a PWR PCV, and consists of a closed cylindrical vessel, 7 m3, with height and diameter of 4.0 m and 1.5 m, respectively; spray condensation tests have been performed in the facility, Porcheron et al., 2007.

3.2.5 ITF for phenomena in containment

3.2.5.1 Scaling considerations related to the PCV-ITF PWR

At the beginning of the PCV analyses, existing facilities, formerly part of NPP, (notably CVTR and HDR) were used for developing an experimental database, and the currently available computational tools were considered suitable for designing of facilities/experiments, and to assess the experimental results. Therefore, scaling analyses received limited attention, NEA/CSNI, 1989a.

For example, the CVTR, Carolinas Virginia Tube Reactor Containment, Schmitt et al., 1970, was designed as the containment for an experimental power reactor. However, facility scaling considerations connected with the design of experiments are not known, NEA/CSNI, 1999. The HDR experiments were performed in the decommissioned "superheated steam reactor" (HeiBdampfreaktor HDR) containment, Muller-Dietsche & Katzenmeier, 1985. Therefore, at the beginning, no specific scaling was considered for the HDR design, Karwat, 1986, and NEA/CSNI, 1989a. The interest on scaling for the PCV-ITF arose when discrepancies were observed between the results from the HDR and the BFC (Battelle-Frankfurt Containment).

In Fig. 3-4 the sketch of three ITFs is shown together with the sketch of pre-stressed-concrete PWR double-containment: reasonably the sketches are reported with the same scale in the horizontal and vertical axes.

Table A3-7 in Appendix A3 shows the main characteristics of the ITFs that can be used for the analyses of PWR PCV, NEA/CSNI1989a.

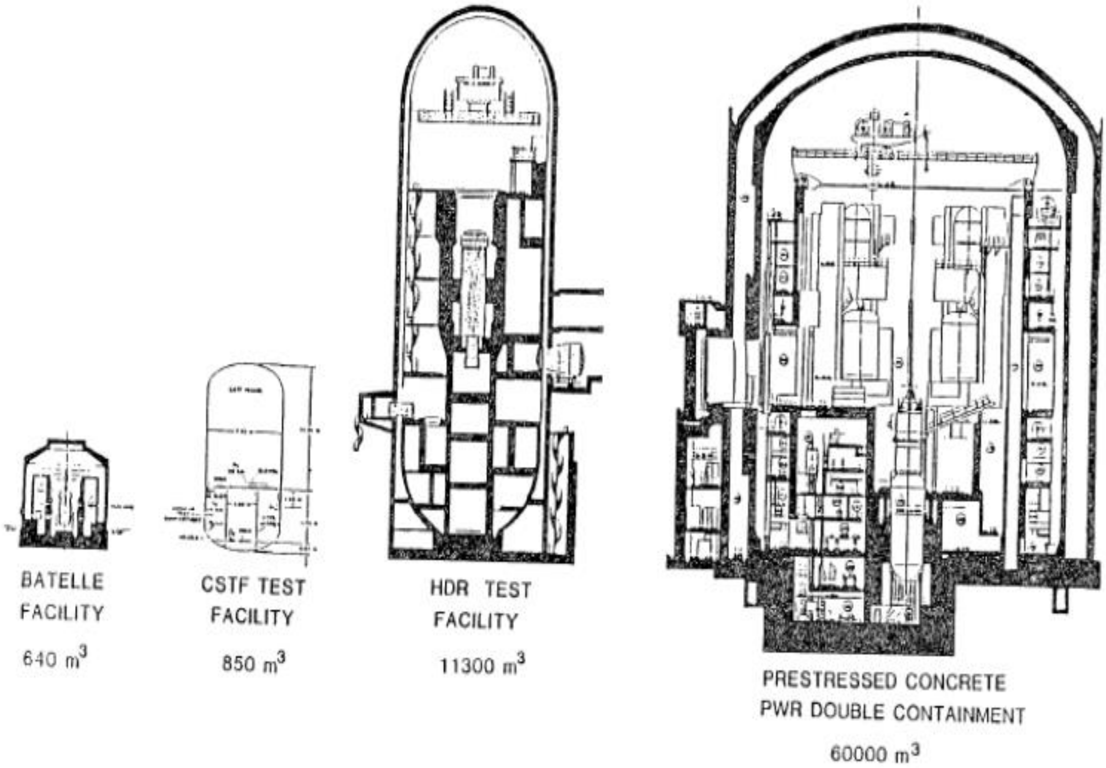


Fig. 3-4 - Comparison of test facilities with pre-stressed concrete PWR double containment, NEA/CSNI, 1989a.

Time Scaling

In the PCV-ITFs, the 'expected time of event' is preserved when the energy- and mass-addition from the reactor primary system and its subsequent distribution along the several compartments of the containment is preserved. This is attained if the mass and energy input is scaled in proportion to the volume scale, NEA/CSNI, 1999. For example, the BFC, Kanzleiter, 1980, was designed to perform experiments on the containment, and the relation of (energy input)/volume ratio was considered in its design, Karwat, 1986.

Volumetric Scaling

The PCV-ITFs listed in Table A3-7 in Appendix A3 have volume ranging from about 1.8 m^3 (Australian Experiment) to about 11300 m^3 (HDR test facility). As in the RCS facility, volumetric scaling determines the ratio (wall surface) / (hydraulic volume) that influences the heat exchange phenomena/process on the wall, NEA/CSNI, 1999. In the concerned conditions, condensation also plays a key role.

Height Scaling

The PCV-ITFs listed in Table A3-7 in Appendix A3 have a height ranging from 3 m (Australian experiment) to 60 m (HDR test facility = prototype height).

Since natural convection takes place in the PCV, it is important to characterize the buoyance-driven convection phenomenon for predicting the thermal-hydraulic response. This phenomenon depends upon the same parameters connected with height scaling of the ITF for the RCS: The (wall surface)/ (hydraulic volume) ratio. An increase in this last parameter causes an increase in the heat exchange between the fluid and confining structures by determining the deposition of energy on to those confining structures. This, in turn, determines the distribution of heat source and sink, NEA/CSNI, 1989a, and NEA/CSNI, 1999.

Material Scaling

Since the thermal conductivity of steel (metal) is high compared with that of concrete, an important factor in facility design that influences the process of heat transfer is constituted by the facility materials, including their relative spatial distribution and proportion. For example, the HDR is a real PCV of a decommissioned reactor, and some steel structures remained in large concrete structures following the reactor decommissioning, Karwat, 1986. The same consideration is applicable to the CVTR where steel structures remained into the concrete. The BFC, instead, consists of a concrete chamber, while steel chamber is used in the Australian experiment, NEA/CSNI, 1989a. THAI, Gupta, 2015, and MISTRA, Studier, 2007, facilities are made of stainless-steel.

The practical influences of CV materials on the response of CV pressure via temperature and/or gas/steam distribution in a huge CV volume, however is not completely understood, e.g. NEA/CSNI – ISP-47, 2007.

As underlined in Table A3-7 in Appendix A3, most of the ITFs considered are characterized by a steel-pressure boundary, while the BFC and CVTR are characterized by a concrete-pressure boundary.

Compartment subdivision and interconnection among compartments

Another characteristic of PCV-ITF is the compartmental subdivision. For example, HDR comprises about seventy (70) compartments, while the BFC can be sub-divided into nine compartments, NEA/CSNI, 1999. However, for the HDR, no exact geometric similarity is present with any other PWR prototype. The same situation is valid for the BFC, NEA/CSNI, 1989a, and Karwat, 1986. It has to be underlined that BFC includes a flow path in a clean way, while the HDR represents a real containment, Karwat, 1986.

Compartment Shape Scaling

The facility shape of all the PCV-ITFs considered in Table A3-7 in Appendix A3 is cylindrical, (NEA/CSNI, 1999). In relation to the shape of some compartments, the domes (ceilings) of the HDR, THAI, and CVTR are hemispherical; otherwise, the same dome has a semi-elliptic shape in the case of CSTF and AP600-PCCS, e.g. see Kennedy et al., 1994; BFC and MISTRA are characterized by a flat dome.

Energy-Release Scaling into PCV

In the ISP-16, ISP-23, and ISP-29, performed in the HDR facility, NEA/CSNI-ISP, 2000, the energy-release rates were scaled to preserve the power/volume ratio expected for full-size PWR. In the CVTR-test facility, the specified blowdown-energy addition was scaled compared with the prototype, NEA/CSNI, 1989a.

Table 3-15 shows some K factors (i.e. scaling factors) defined for the PCV-ITF-PWR facilities. The definition of the K factor is given by D'Auria & Galassi, 2010.

$$K_{\varphi} = \frac{\varphi\ quantity\ value\ in\ the\ experimental\ test\ facility}{\varphi\ \ quantity\ value\ in\ the\ reactor\ prototype}$$

Table 3-15 - Main PCV-ITF facility, applicable to PWR design, scaling factor summary.

FACILITY	KP	KV	KH	KD	KDV	KC	KS	KCON	KTOTS	KS/C
HDR	1.13	0.16	1.07	0.37	0.12	0.58	1.45	0.27	0.65	4.00
BFC	0.94	0.01	0.16	0.21	0.01	0.075	0.01	0.03	0.025	2.72
CVTR	0.28	0.09	0.62	0.31		0.025	0.17	0.04	0.08	0.89
Australian experiment		0.00003	0.05	0.02		0.02	0.001	0	0.0003	11.80
CSTF	0.94	0.01	0.36	0.14	0.02	0.02				
AP600 PCCS	1.3	0.0013	0.11	0.08	0.002	0.03				
MISTRA	1.13	0.0014	0.13	0.08						
THAI	2.64	0.0009	0.16	0.06	0.0004	0.04	0.011	0	0.0036	4.2
German PWR Containment	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00

The parameters listed in the table are from Mascari & De Rosa, 2015:

- ***P (KP): Containment Pressure***
- ***V (KV): Containment Total Volume***
- ***H (KH): Containment Height***
- ***D (KD): Containment Diameter***
- ***DV (KDV): Containment Dome Volume***
- ***C (KC): Containment Compartment Number***
- ***S (KS): Containment Steel Surface***
- ***CON (KCON):Containment Concrete Surface***
- ***TOTS (KTOTS): Containment Total Surface***
- ***S/V (KS/C): Containment Surface / Volume ratio***

In Table 3-16, the PCV ISPs within the DBA envelope are identified and the relevant scaling information is provided.

Table 3-16 – Containment ISP relevant to DBA and scaling information.

ISP	Type	ISP Title	Facility	Relevant scaling consideration	DBA	SA/ BDBA
CASP1	INT	Steam-line Rupture within a Chain of Compartments (Battelle Test D15)	Battelle model containment	• elevation scaling approximately 1/5 related to PWR containments • volume reduction related to commercial PWR containments approximately 1/100 • internal surface/volume ratio approximately 2.4 (as compared to approximately 0.9 surface/volume ratio of commercial PWR containments)	YES	yes
CASP2	INT	Water Line Rupture into a Branched Compartment Chain (Battelle Test D16)	Battelle model containment	• elevation scaling approximately 1/5 related to commercial PWR containments • volume reduction related to commercial PWR containments approximately 1/100 • internal surface/volume ratio approximately 2.4 (as compared to surface/volume ratio approximately 0.9 of commercial PWR containments)	YES	Yes
CASP 3	INT	Small-Scale Two-Compartments Basic Containment Experiment	Lucas Heights blowdown/ containment test rig	• internal surface/volume ratio approximately 6.1 (as compared to commercial PWR containments approximately 0.9)	YES	Yes
16	INT	Rupture of a Steam Line within in the HDR Containment Leading to an Early Two-Phase Flow	HDR	• large scale test facility, cylindrical shaped steel shell containment, typicality of compartment arrangements; geometrical similarity not preserved; • volume scaling compared to full-size PWR approximately 1/6 • energy release rates scaled to preserve power/volume ratio expected for full size PWR; • time preserving of mass and energy release rates resulting in preservation of typical pressurization transients • "as measured" mass and energy-release rates cross-checked by supplementary blowdown calculations	YES	NO
17	COM	Marviken: Pressure Suppression Containment-Blowdown Experiment No.18	Marviken	Commercial containment:, however, no similarities to BWR typical pressure suppression system containments	YES	YES
23	INT	Rupture of a Large-Diameter Pipe in the HDR Containment	HDR	• large scale test facility, cylindrical shaped steel shell containment, typicality of compartment arrangements; geometrical similarity not preserved; • energy release rates scaled to power/volume ratio; • time preserving of mass and energy release rates results in typical pressurization transients • "as measured" mass and energy release rates cross-checked by supplementary blowdown calculation	YES	YES

ISP	Type	ISP Title	Facility	Relevant scaling consideration	DBA	SA/ BDBA
29	INT	Distribution of Hydrogen within the HDR Containment under Severe Accident Conditions	HDR	• large scale test facility, cylindrical shaped steel shell containment, typical of compartment arrangements; however: geometrical similarity not preserved; • energy release rates scaled to preserved power/volume ratio; • time- preserving of mass and energy release rates results in typical energy-discharge transient • as measured mass- and energy-release rates cross-checked by blowdown calculation • hydrogen release into the containment simulated by release of a helium/hydrogen mixture, rates scaled proportional to the reduced containment volume • hydrogen release initiated 740 min after begin of SBLOCA transient	YES	YES
37	INT	VANAM M3-A Multi Compartment Aerosol Depletion Test with Hygroscopic Aerosol Material	Battelle model containment	• volume- scaling related to commercial PWR containments approximately 1/100 • elevation reduction, approximately 1/5 related to commercial PWR containments • internal surface/volume ratio approximately 2.4 (as compared to approximately 0.9 ratio of commercial PWR containments) • boundary conditions of VANAM test not typically expected for severe accident situations in real plants	YES	YES
41	COM	RTF Experiment on Iodine Behaviour in Containment Under Severe Accident Conditions	Radioiodine Test Facility (RTF)	Fundamental experiment not directly scaled to assumed severe- accident conditions of NPPs	YES	YES
42	INT	PANDA Test "TEPPS"	PANDA	integral systems experiment, but without scale-relations to existing nuclear power plants basic testing of new containment features	Phase A: YES, Phase C: YES, Phase E: N/A, Phase F: N/A	Phase A: NO, Phase C: NO, Phase E: YES, Phase F: YES
47	MISTRA: COM, THAI: INT, TOSQAN: COM	ISP-47 ON CONTAINMENT THERMAL HYDRAULICS	MISTRA, THAI, TOSQAN	Volumetric scaling is addressed in TOSQAN and MISTRA experiments by running similar test sequences (steady-state conditions with the same gaseous mixtures). Conservation of dimensionless numbers (Grashof, Richardson…) is usually required to achieve similarity, but the test sequences in the three facilities have not been designed to respect such criteria.	MISTRA: YES THAI: YES TOSQAN: YES	MISTRA: YES THAI: YES TOSQAN: YES
49	SE	Hydrogen Combustion	THAI, ENACEEF	-	THAI: YES ENACEEF: NO	THAI: YES ENACCEF :YES

3.2.5.2 Advanced reactor design considerations

In the current reactor design, it is possible to study the PCV physical behaviour separately from the RCS-physical behaviour, NEA/CSNI, 1996b. The RCS is the source of water/steam and hydrogen for the PCV, and can be considered as a boundary condition for the PCV analyses, NEA/CSNI, 1999. In the new advanced passive-reactor designs, it is not possible to consider the RCS as a boundary condition for the PCV, but it is necessary to consider the physical behaviour of the PCV coupled with the RCS physical behaviour, NEA/CSNI, 1996b. It also is necessary to characterize the RCS/PCV coupled behaviour during the evolution of the transient. This requirement is due to the strong coupling effects and feedbacks between the RCS and PCV. The passive mitigation strategy depends on the characteristics of natural circulation loop that are in response to both components (PCV and RCS) to remove decay heat. Two PWR examples can be the AP600 and MASLWR concerning their SBLOCA passive-mitigation strategy.

In the OSU-MASLWR facility, for example, two vessels – High Pressure Containment (HPC) and Cooling Pool Vessel (CPV) - thermally connected by a heat structure (Heat Transfer Plate) are used to simulate, respectively, the prototypical containment structure, wherein the RPV (primary vessel) resides, and the pool where the containment structure is located. The Heat Transfer Plate simulates the proper heat-transfer area between the containment structure and the pool. Since, during a blowdown experiment in the scaled-down facility, the steam coming from the RPV has to be condensed along the Heat Transfer Plate, heaters are installed in the shell wall of the HPC, Modro et al., 2003, and Reyes et al., 2007.

An International Collaborative Standard Problem (ICSP) on 'Integral PWR Design, Natural Circulation Flow Stability and Thermo- hydraulic Coupling of Primary System and Containment during Accidents', was conducted in the OSU-MASLWR facility to assess computer codes for designing reactor systems and for their safety analysis, IAEA, 2014.

An example of the behaviour of an advanced BWR containment coupled with the primary side is present in the SBWR design. As highlighted in the previous Section, the major integral test programs related to the SBWR have been conducted at the GIST, GIRAFFE, PANDA, Gamble & Fanning, 2006, and PUMA ITF, J NED Special Issue, 1998 (Ishii paper).

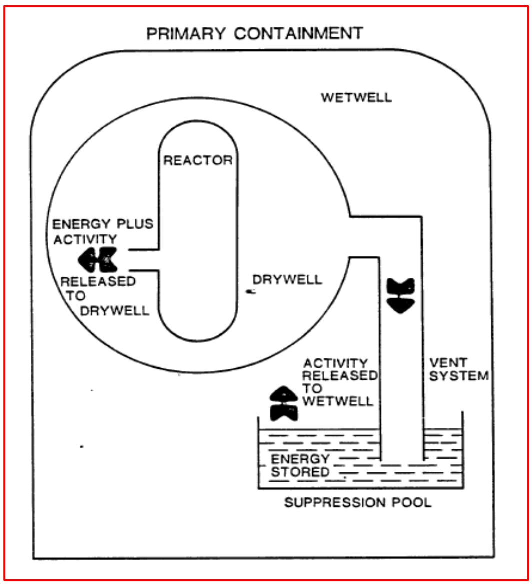
The Panda facility, for example, was designed to simulate the containment of the SBWR with a volumetric scale of 1: 25, and a height scale of 1: 1, IAEA, 1995a. It is noted that in the PANDA test facility, different programs have been performed as reported by Paladino & Dreier, 2012, for example: The Simplified Boiling Water Reactor (SBWR); The Economic Simplified Boiling Water Reactor (ESBWR); and the SWR-1000 with a Building Condenser. The CSNI ISP-42 has been performed in the facility and a counterpart test in PUMA facility is reported by Yang et al., 2013.

Another advanced BWR is the KERENA reactor. The INKA facility is a full-height, volumetric-scaled test facility (1: 24 in volume) of the KERENA containment aiming at characterizing the performance of the passive safety-systems of KERENA, Leyer & Wich, 2012. The test facility models the RPV and all containment compartments, including the pools. It has all interfaces for transferring heat and mass between the RPV and drywell, the drywell and wetwell, as well as between the drywell and a shielding and storage pool outside the containment. The INKA test facility realistically features the KERENA functions of pressure suppression and core flooding, all fully passive, i.e. without active safety-systems.

3.2.5.3 Brief analyses of the scaling problems of the BWR containment test facility

BWR containment designs are characterized by a "pressure suppression containment" being constituted with "wet well" that has a large suppression (water) pool covered by gas-space above it, a "dry well" that is a pressure-retaining structure surrounding the RCS, and a "vent system" that connects the dry well gas space to the wet well below the surface of the suppression pool water surface, NEA/CSNI, 1986.

A schematic view and the essential features of the pressure-suppression containment are given in Figs. 3-5 and 3-6. Several and different kinds of pressure-suppression containments have been designed and used. The main key component data are shown in Table A3-8 in Appendix A3. A typical layout of BWR GE-MARK I, II, III configuration is shown in Fig. 3-6.



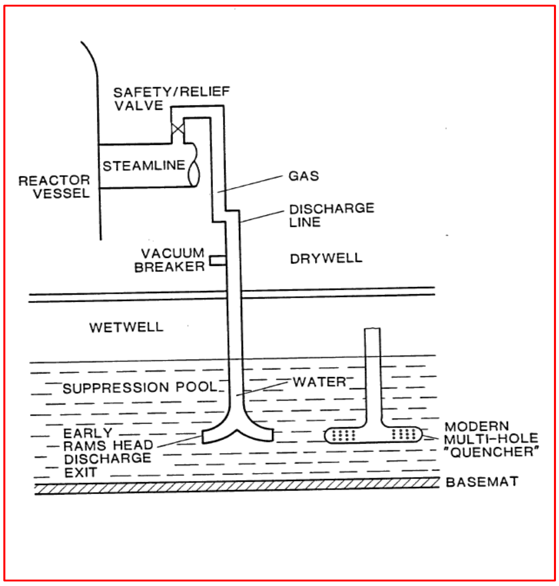


Fig. 3-5 – Left: Schematic of essential features of a pressure-suppression containment;
Right: Typical Safety/relief valve-discharge line, NEA/CSNI, 1986.

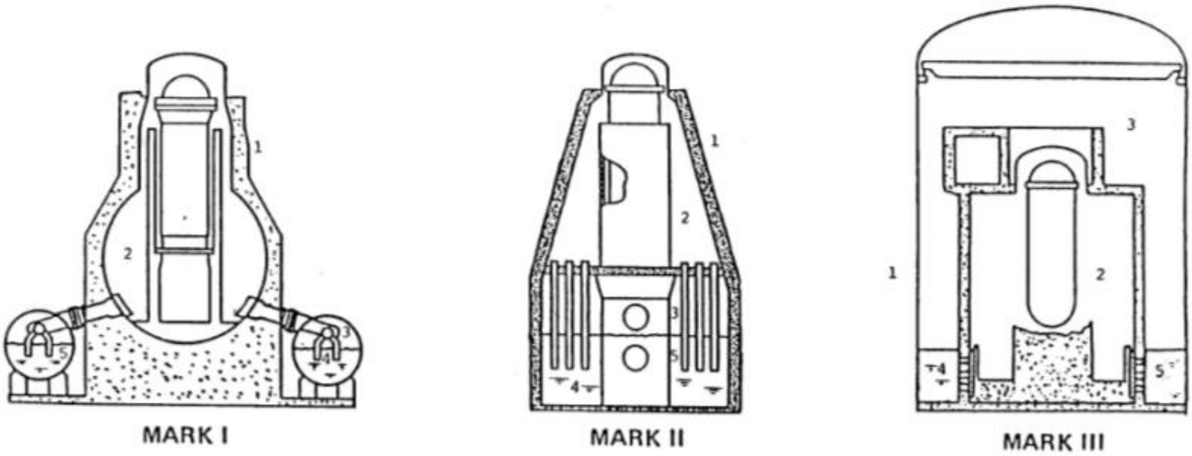


Fig. 3-6 – GE pressure suppression system design, NEA/CSNI, 1986.
(1 = primary containment; 2 = dry-well; 3 = wet-well; 4 = PSP; 5 = vent system)

One of main features of the BWR containment is the large capacity suppression pool, acting as a heat sink. The phenomena of interest associated with the pressure suppression pool take place during the safety/relief valve blowdown, typically during ADS actuation and blowdown during the LOCA. In relation to the “safety/relief valve blowdown tests” it is to highlight the fact that a KWU series of scaled- and full-scale-prototype discharge-device tests have been executed to reduce short-term line clearing load and to improve the pool operating margins. In-Plant tests were performed to confirm the extrapolation of the results, NEA/CSNI, 1986.

In relation to the LOCA- related tests, some of the BWR containment scaling topics and requirements are described in detail in NEA/CSNI, 1986, which, of particular interest is the "single cell" hypothesis. The dynamic similarity is obtained if the volumes of the drywell, the wet well, and water in the suppression pool are in proportional to the corresponding prototypical volume divided by the prototypical number of vent pipes. This should be coupled with a full-vent pipe dimension, representative pool surface area, and geometric volumes. Other important points are the specific energy additions, fluid-structure interactions, and the simultaneous interactions of a large number of vent pipes. The rate of specific-energy additions rate per vent pipe has to be preserved to assure dynamic similarity. The confinement of the wet well pool was, in general designed with an eigen-frequency behaviour similar or identical to that of the prototype plant of interest so to assure the warranted proper fluid-structure interaction. Simultaneous interaction of a large number of vent-pipes is of interest for the loads on the prototype structures caused by oscillations in condensation and steam chugging. In relation to this subject, seven full-size vent pipes were used to simulate the vent-system response in the JAERI full scale MARK-II containment response test program, Kukita et al., 1984. In relation to the fluid structure interaction and the overall system response, great care was taken to assure for the correct distribution of structural masses of the entire system. A similar approach has been used for the design of the FSTF facility for the MARK-I containment, NEA/CSNI, 1986.

3.2.6 Scaling distortions in experiments

Major reasons for scaling distortions

All integral tests and most of the separate effect tests were conducted in scaled-down test facilities that have unavoidable scaling distortions. Consequently, the effects of the scale and scaling distortions on processes and/or parameters of interests must be assessed.

As mentioned in Section 2.2.1, it is well recognized that a complete similitude cannot be achieved, particularly for a complex nuclear reactor system. Therefore, it is important to prioritize the similarity of the processes of the greatest interest between the prototype and the model. The assumptions and compromises made in the scaling method may distort less-important processes, Zuber et al., 1998. Thus, scaling distortions are inevitable in scaled-down test facilities, and compromises occur due to the difficulty of attaining complete similitude in all local phenomena, and our lack of knowledge of the local phenomena themselves. The conflict of scaling criteria imposed by different physical phenomena and processes leads to scaling distortions. Also, some scaling distortions in the test facility are due to the limitation of the engineering construction of the facility.

A conflict of scaling criteria imposed by different physical phenomena and processes usually involves the priority of the scaling criteria, and leads to scaling distortions. Distortions also exist in the test facility due to the limitation of engineering manufacturability and the operability of the facility.

The scaling distortions do not have the same effect throughout a transient. They may not affect transient behaviours in some phases of the transient, while they do in other phases of the transients, Boyack et al., 1989. The effects of scaling distortions change as transients proceed through its phases. These distortions by test facilities, as well as the initial- and boundary-conditions should not affect the evolution of major physical processes.

A facility designed for a particular type of postulated accident scenario may have significant scaling distortions when the facility is used to simulate the different types of accident scenario.

Thus, it is common approach to optimize the similitude for processes of greatest interest. This may lead to a similitude with distortions of other lesser important processes in a scaled-down test facility. It is necessary to evaluate their effects and/or of distortions of the test initial- and boundary-conditions on the evolution of a transient.

Scaling distortions in Circular Sections

For a scaled-down test facility with a reduced diameter scaling (power/volume with full height), the hydraulic diameters in circular sections, such as pipes and downcomers, are not preserved, Larson et al., 1980, because of difficulties in directly applying the system global-scaling factor. Regardless of the scaling laws used in the design and the construction of the test facility, all scaled-down test facilities may have the following scaling distortions. First, the drop in frictional pressure in the test facility may become large due to an increase in the length-to-diameter ratio. Second, the transfer of structural heat may be distorted by the increased ratio of the surface area to fluid volume. Finally, some local phenomena, such as flow regime development/transition, and the CCFL may be distorted due to the small hydraulic diameter in the test facility. In the Semiscale mod-3 facility, two-phase flow separation was promoted in the oversized horizontal pipes. The hydraulic diameters of the hot- and cold-legs and downcomers are optimized to reduce the scaling distortions in the scaled test facility. The various aspects and effects of the scaling distortions are explained below further.

Scaling distortions by a structural-heat loss and stored heat

Scaling distortions often occur in the heat losses and the stored heat of the facility structures, Nayak et al., 1998. The structural scaling distortions in the scaled-down facility increase due to a larger structural mass, and structure surface area per unit coolant volume relative to the prototype. The resulted scaling distortions in the facility structure cause excessive heat transfer to the fluid and a large heat-loss to the environment, depending on the type of transient Boyack et al., 1989, Zuber et al., 1990, and NEA/CSNI, 2001. For a given circular geometry such as a pipe, the ratio of surface area to the fluid volume is given by 4/D (where D is the pipe diameter). Obviously, a small-diameter pipe will have a much larger surface area to the fluid volume relative to the prototype. It is noted that structural mass per unit coolant-volume is reduced in low pressure test facilities (due to their thinner walls), and typically is closer to the prototype value compared to that in high-pressure test facilities.

Piping connections, such as flanges and nozzles and even instrument cables, may contribute further to the heat loss. Scaling distortions, induced by the heat loss, generally affect relatively slow (or long-term) transients, such as a SBLOCA. The increased heat loss decreases the fluid temperature and system pressure and, consequently, safety injection is actuated earlier than the time expected in the prototype. Also, since the increased condensation of steam on the wall affects the void fraction and the flow regime, the flow rate at the break and the coolant inventory may be changed through the influenced response to system pressure. The scaling distortions by the heat losses may also affect heat transfer to the SG secondary side.

The stored energy in the scaled-down facility increases due to a larger structural mass- to- volume ratio than that of the prototype. Consequently, the time constant for releasing the stored energy to fluid is reduced due to the small thickness and increased area of heat transfer. Scaling distortion of the stored energy significantly affects the transients of both the LBLOCA and the SBLOCA. In the LBLOCA, the release of stored energy significantly can change the initial conditions, and, as a result, a significant scaling distortion exists during the period from re-flood phase to the long-term cooling phase. This large release of stored energy increases the fluid temperature through excessive heat transfer, and the system pressure can be maintained higher than that of the prototype. In summary, the effect of the large stored energy is opposite to that of the heat loss.

In general, a tall thin test facility has overwhelming heat-transfer to the structure, a feature that has been the major shortcoming of the smallest integral test-facilities. Spatial and time distributions of the heat losses may affect the structural scaling distortions. The scaling distortions due to the heat loss can be compensated for using countermeasures, such as a tape heating system. Initial operating conditions can be modified slightly to reduce the scaling distortions due to the stored heat.

The scaling distortions related to a structural heat transfer have the most influential effects on the system depressurization in small breaks, Karwat, 1985. A different system depressurization affects the break flow, fluid conditions, and the initiation of timing of the emergency core-cooling flow. According to the sensitivity analysis of overall system behaviour of SEMISCALE-mod-3 due to heat loss using the RELAP4 system code, the heat loss was predicted to influence greatly the system depressurization by maintaining lower system pressure in general, and resulting in the early actuation of the accumulator and the low-pressure safety system, Larson et al., 1980. The core power can be increased to compensate for the large heat losses due to atypical larger surface area-to-fluid volume ratios. The increased core power may affect local phenomena, such as the increase of the void fractions in the core. Thus, guard heaters on the system were installed to compensate for the structural heat loss in the SEMISCALE-mod-3 facility.

Scaling distortions related to the inventories and inter-component flows

For a system consisting of several inter-connected components, a proper scaling of the inter-component relationships is important to preserving the thermal-hydraulic interactions among these components, Ishii et al., 1998. If the system pressure of the components is prototypical, all thermal-hydraulic properties of the liquid, vapor and liquid-vapor mixtures will be preserved in the scaled-down facility. The scaled mass and energy inventories for each component can be obtained from the control volume balance equations for mass and energy. At the interface between two connected components, the scaling criteria are obtained from the inter-components' mass and energy flows. The discharge flow phenomena at the breaks, and at the safety and depressurization valves, should be preserved so to assure similar depressurization histories between the prototype and the scaled-down facility. To preserve overall transient behaviour between the prototype and the scaled-down facility, the depressurization histories should be the same compared to their respectively scaled time-frames.

For the similarity of the depressurization histories, the component coolant mass, divided by the component volume, should be preserved in the scaled-down facility. A separate scaling criterion for the system boundary flows, such as the break flow, and various ECCS injection flows, can be obtained to preserve the dimensionless mass-conservation equation. The scaling criteria for energy flow also can be obtained from the dimensionless energy-equation. For full-pressure simulation, the inflow and outflow should have prototypic enthalpies, and the initial energy inventories also should be scaled by the volume ratio.

The pressure difference for inter-component flow should be prototypic if the flow mainly occurs due to the pressure difference. However, in gravity-driven or natural-circulation flow, the hydrostatic head is the main driving force. Then, the pressure difference should be scaled down by the height ratio.

A detailed scaling analysis for the inter-component flows is given in the report by Ishii et al., 1996.

Scaling distortions related to the pressure drop

It is important to preserve the similitude of the distributions of pressure drops over a test facility, which determines the flow distributions along the flow paths. The pressure distributions along the primary- and secondary loops of the scaled-down facility should be the same as those expected in the prototype as an essential condition to preserve the response of natural circulation, for example.

If a scaled-down facility is tall and thin, the length divided by diameter is very large. Then, the pressure drop (or hydraulic resistance) along the loop piping will be distorted in the facility. Compromises usually are necessary to assure the similarity of pressure drop in the scaled-down facility. For example, loop configurations are changed to preserve the pressure drops typical of the prototype. While maintaining the scaled fluid volume in the loop piping, the similitude of the loop pressure drop is preserved by using a pipe having an oversized diameter and a shortened length.

The primary piping diameters of the SEMISCALE-mod-3 (volume scale of 1/1705 with full height) and the MIST (volume scale of 1/817 with full height) facilities are enlarged compared with the ideally scaled values to preserve the similarity in the pressure drop, Patton, 1978, Larson et al., 1980, and Young & Sursock, 1987. The increase of the pipe diameter distorts the local fluid velocity. The oversized primary piping may affect thermal-hydraulic phenomena, such as phase separation, flow-regime transition, counter- current flow limitation, and entrainments. The influence of these phenomena on break flow and the general system responses need to be assessed to quantify and minimize scaling distortions.

Scaling distortions related to multi-dimensional phenomena

If a scaled-down facility is tall and thin, physical phenomena can be distorted in the upper head, the upper- and lower-plena, the reactor vessel downcomer and the SG plena. The RPV downcomer in the scaled-down facility will have a narrow gap and a very large surface area to fluid volume ratio. This will affect phenomena, such as flow pattern, steam condensation, Emergency Core Coolant (ECC) bypass, mixing, liquid entrainment and de-entrainment in the upper plenum, liquid carry-over to the steam generators causing steam binding, which are important during the refill and reflood periods of a LB LOCA, Zuber et al., 1990.

Most currently available scaling laws have been developed based on the one-dimensional approach. This feature induces practical limitations on their application to designing and operating of the scaled-down experimental facility whose main objectives is to simulate the multi-dimensional phenomena, Song, 2006. These phenomena are easily observed in large components of nuclear power plants, such as RPV and SG; the ECC bypass phenomena during the refill phase of LBLOCA, boron mixing in the RPV, downcomer boiling, and thermal mixing/stratification in the hot and cold legs. Passive (gravity-driven) safety designs may cause significant multi-dimensional phenomena because of the weak driving force.

While the system TH codes for existing systems mostly are composed of one-dimensional models, they should be validated against the multi-dimensional phenomena that may appear, especially in advanced reactor designs. In addition, the existing scaling laws should be checked against the multi-dimensional phenomena.

As such, it is essential to assess the predictive capabilities of the existing system analysis codes for multi-dimensional phenomena.

Song et al., 2006, detailed the scaling of multi-dimensional thermal-hydraulic phenomena. They analysed the scaling and similarity of experimental data on the ECC bypass phenomena during PWR LBLOCA refill phase by applying the modified linear scaling method. The scaling method preserved an aspect ratio, and reduced the velocity of flow in a scaled-down facility for preserving the ECC bypass phenomena. These phenomena in a full-scale facility (UPTF) were compared with those in several scaled-down test facilities built according to the modified linear-scaling method. The liquid film spreading width and the direct ECC bypass fraction showed good similarities, regardless of the scales of the test facilities. Thus, the modified linear-scaling method preserved the multi-dimensional behaviours in the RPV downcomer during the LBLOCA refill phase.

Scaling distortions due to a scaled-down reactor-coolant pump

The behaviour of reactor coolant pump in a PWR may significantly influence the system behaviour and the distributions of the coolant inventory during a postulated accident, Choi et al., 2008. The timing of the trip of the reactor coolant pump affects the coolant inventory in the primary system. During the blowdown phase of LOCA, the system pressure decreases rapidly, and the reactor coolant reaches saturated conditions. Then, a two-phase mixture of water and steam circulates through the primary coolant system and the reactor coolant pumps. It thus affects the boundary conditions for subsequent refill and reflood phases of a LOCA. However, it is very difficult to predict the pump behaviour in the progression of the accident scenarios due to the complex interaction between the fluid in motion and the rotating impeller. Namely, the pressure drop across the RCP in the intact loops or in the broken loop, in parallel to a path through the core, may affect the location of stagnation point during blowdown (following a LBLOCA) and, therefore, may have a large impact upon the PCT (e.g. when the stagnation point coincides with the highest location of core power). The most important behaviour of RCP is a two-phase flow performance since it affects significantly the coolant distribution in the primary loop, during the blowdown and refill phases. In the late-phase LBLOCA pressure drops across the pump may affect ECC mass loss from the break, depending on the (assumption of) the break location.

The effects of the pump operation on thermal-hydraulic behaviour during a small break LOCA were investigated in the SEMISCALE-Mod-3 integral test facility, Johnson, 1980. The system hydraulic behaviours depend on the pump operation scenarios, i.e. pump trip at scram, delayed trip, and continuous pump operation.

Kamath & Swift, 1982, suggested that pumps of different designs experience similar two-phase effects as long as their specific speeds and single-phase characteristics are similar. Pump models will affect the initial- and boundary-conditions in the reflood and refill processes of a LOCA. Though several experimental- and analytical-works were carried out with various pumps under a wide range of conditions, a reliable two-phase pump model is not available until now. Thus, it is usual to carry out experiments to determine the pump behaviour in two-phase flows, as well as in single-phase flows.

By a dimensional analysis, i.e. the Buckingham Pi theory, it is possible to obtain three major dimensionless parameters, i.e. specific speed, specific capacity, and specific head, governing the pump performance. For the pump flow to be dynamically similar, the three dimensionless parameters should be preserved in the scaled-down facility. However, it practically is impossible to preserve these three dimensionless parameters. For example, a scaled-down facility requires pumps with very large rotational speed so to preserve the specific speed.

Ideally, the similarity of the reactor coolant pumps in the scaled-down test facility requires the following scaling criteria. The type of the reactor coolant pumps (i.e. radial-, mixed-, or axial-flow types) should be preserved. The geometrical similarities, i.e. dimensions, clearance, and angles, should be preserved in the scaled-down facility. Other geometrical similarities include the number of pump blades, and the angles of the inlet and outlet blades. The diameter of the pump impeller should be designed according to the scale ratio of a diameter in the scaled-down facility. The specific speeds of the pump should be preserved in the scaled-down facility. If it is not preserved, proper measures should be taken in the scaled-down experiments. In this case, one of the general methods is to control the pump speed according to each accident scenario. It can be assumed that the cavitation characteristics of the pumps are preserved when the specific speed is well preserved. Also, the head, mass flow rate, pump inertia, and torque should be determined by similarity criteria. Normalized homologous curves in the scaled pumps should be preserved in the scaled-down facility. The coast-down characteristics and flow resistance of the scaled pumps should be similar to the prototype. To assess a scaling distortion of the scaled pumps, it is general to perform scoping analyses for various accident scenarios using system analysis codes, Annunziato, 1985.

Scaling distortion by fuel simulators

Electrically heated fuel simulators may behave very differently from nuclear-fuel rods, specifically in a LBLOCA, Grandjean, 2006. SEMISCALE solid-type electrical heaters are composed of a spirally wound heating element embedded in the ceramic insulator show different behaviours with nuclear fuel- rods due to their different thermal properties, and lack of a fuel-pellet cladding gap, Brittain & Aksan, 1990. In-pile tests carried out in the Halden reactor revealed a significant delay in quenching for the solid-type electrical heaters compared to nuclear-fuel rods; on the other hand, the REBEKA type electrical heaters showed similar behaviour to that of the nuclear fuel rods. The nuclear fuel rod with a gap was calculated to cool approximately five times faster than the SEMISCALE sold-type heater rod, Brittain & Aksan, 1990. The thermal decoupling of the cladding and fuel pellets due to the gap was significant, allowing the cladding to quench rapidly during the blowdown phase.

In addition, when using electrically heated fuel simulators it can be difficult to simulate a fast feedback phenomenon due to the coupling of neutron physics and thermal-hydraulics, such as the effect of void reactivity on the natural stability of coolant circulation.

Another aspect related to fuel simulation is the influences of radiation (gamma-ray). Because of quite strong gamma-ray generation in the core, all the material surfaces in the core become super (ultra) hydrophobic. While experimental evidence is limited, this phenomenon causes the propensity of the surface to re-wetting as well as increase in the critical heat flux (CHF) by delaying departure from nucleate boiling (DNB), Sibamoto et al., 2007, especially in low pressures during re-flooding in the PWR LBLOCA. This is difficult to represent in the experiment facilities without irradiation capability for the whole height of simulated fuels.

Scaling distortions of local phenomena

Scaling distortions can be encountered due to atypical local phenomena in a scaled-down facility. These distortions occur due to inherent scaling distortions by design and simulation constraints, and to the non-typicality of local phenomena.

For example, two-phase flow regimes are dependent on pipe diameter, as partly discussed in Section 3.1. The slug flow regime can no longer exist in vertical flows when the pipe diameter is large (large pipe), provided that the pipe diameter is far larger than the size of the cap bubble. According to Laplace length scale, a large diameter pipe is defined as a pipe in which the dimensionless diameter defined by Eq. (3-64) is greater than about 32-40, Boucher et al., 1990, as also discussed in Section 3.1.3. When it is larger, the structure and dynamics of vertical two-phase flow may dramatically change Schlegel et al., 2009, and Hibiki & Ishii, 2003. Thus, vertical two-phase flows in a small scale test facility can be quite different from those that may appear in the full-scale prototype. A similar argument should be true for the change in phenomena under high-temperature; thus under high-pressure, conditions such as bubble size become small when the surface tension is decreased. Partly because of this, demonstrating an experiment under prototypical pressure, temperature, and mass flux is necessary to observe DNB and to define the CHF usable for analysing reactor safety.

Another example of scaling distortion is the impossibility to scale-up vapor formation by sharp-edge cavitation. This may affect the TPCF downstream. This phenomenon depends on geometric parameters around the break.

Some two-phase flow phenomena strongly depend on scale, such as heterogeneous steam-water counter-current flow limitation, Glaeser & Karwat, 1993. UPTF tests showed significant scaling-dependent experimental results in the case of heterogeneous steam-water flow conditions. The scaling dependence mainly was due to a pronounced multi-dimensional two-phase flow that is difficult to observe in a small-scale test facility.

Multi-dimensional behaviours, like thermal stratification and natural convection occur in various components with liquid in the tank, such as suppression pools, safety injection tanks, and even in PWR horizontal legs. The thermal stratification in the suppression pool will impact the rate of heat-transfer in the heat exchanger installed in the suppression pool. The condensation rates in a tank also are influenced by the degree of the thermal stratification. Such effects in horizontal pipes are not expected to be strong if the pipe diameter is narrow. Thus, the thermal stratification effects in the full-size prototype are expected to be stronger than those in a small test facility. Multi-dimensional natural convection in a tank also depends on the geometrical sizes. System codes using one-dimensional approach will have difficulty in predicting the effects of scaling distortion in the thermal stratification on the transient behaviour. Thus, computational fluid dynamics code with the system codes can be helpful to take into account the effects of thermal stratification on the transient behaviour.

Phase separation significantly influences heat transfer and the quenching of core. Water holdup in pressurizer can be affected by the efficiency of phase separation upstream of the surge-line inlet. The efficiency may depend on the surge line' design as well as on the facility scale, Kukita et al., 1990, and Liu et al., 1998.

3.3 Counterpart Testing

Experiments at ITFs provide a substantial contribution to the resolution of safety issues of NPPs and the understanding of an NPP behaviour under off-normal conditions. As the typicality of the experimental data acquired in experiments at a single (scaled) test facility may be questioned in some cases due to inherent scaling distortions resulting from construction compromises and simulation constraints, the concept to conduct a counterpart test campaign involving several ITFs at different scales and design concepts were realized with decision to construct the first ITF (e.g. Semiscale/LOFT, see also D'Auria & Galassi, 2010).

Such experimental efforts are considered highly beneficial, not only for analysing a LWR thermal hydraulics independent from computational analysis, but also to demonstrate the adequacy of system codes in predicting a realistic system response, and to assessing uncertainties of calculation models.

One of the first and most comprehensive counterpart test campaign on an SB-LOCA scenario covering six experiments from four ITFs was conducted more than 20 years ago, together with all the accompanying analytical work and derived procedures, as a reference for such experimental campaigns. Even now, in, the concept to conduct counterpart experiments still is considered to be of high value for solving of current NPP-related safety issues. This CT programme is under consideration by the growing international collaboration, partly within international programs with participants from various organizations including operators, licensing authorities, licensing expert organizations, research institutes, and manufacturers.

It is useful to distinguish between similar tests and counterpart tests. The following sub-Sections elaborate on the definitions for these two kinds of experiments in more detail. In addition a few examples for the beneficial utilization will be given, as well as the limitations of the concept under scaling aspects.

3.3.1 Counterpart Tests and Similar Tests

The words, Counterpart Test (or Tests), CT, and Similar Test (or Tests), ST, are tightly connected with the technology of ITF built for simulating transient- and steady-state conditions in NPPs. Several tens of ITFs have been built and operated to simulate PWRs and BWRs all over the world in the last fifty years. Several hundred (summing up to more than 1500) 'integral' experiments were designed and performed. Each experiment has a duration ranging between a few tens of seconds (e.g. LBLOCA) and several days (e.g. transient scenarios where passive systems or accident management procedures play a role); each experiment is characterized by a number of recorded variable trends ranging between 50 and around 10,000. Dedicated literature exists, e.g. NEA/CSNI, 1996c, NEA/CSNI, 1996d, and D'Auria, 2001, and it is not the purpose of the S-SOAR to duplicate the related information. However, to arrive at the definitions of CT and ST, and to propose a meaningful use of the related data, essential ITF information is provided below and in Appendix A3.

Most of the ITFs have been designed and built according to the following rough scaling-principles, laws or scaling parameter values, in each case related to the respective prototype NPP:

- power-to-volume (Kv) is kept, and preserving the time sequence of events is a target;
- full pressure is kept for initial conditions, such that preserving the pressure evolution during the transient is a target;
- full linear power (even though in the majority of cases this objective has been achieved only in relation to decay power) is kept as a boundary condition primarily;
- pressure drops (including those at geometric discontinuities) and fluid temperature/void distributions in the RCS are kept for initial conditions, and their preservation during a transient constitutes a target; this also implies constraints on the design of recirculation pumps;
- full height of key components is kept;
- number of loops, configuration of downcomer (see also Section 2.1.3.1), lengths and diameter of hot- and cold-legs typically constitute a matter of designer choice;
- Elements like the geometric configuration of the pressurizer surge line, geometric- and

material-configuration of fuel rods (see also Section 2.1.3.6), configuration of different flow bypasses inside the RPV, connection between ECCS and main pipe in the RCS and key valves operating conditions, also constitute the matter of designer choices.

Definitions for CT and ST already are a part of documents agreed by the international community, e.g. NEA/CSNI, 1996d. In the case of CT, rigid definitions imply:

– same prototype NPP for the ITF involved in the CT;
– same scaling concepts for any of the bullet items above;
– Boundary- and initial-conditions (of CT experiments) are properly scaled (i.e. according to a set of scaling principles accepted by each group of researchers managing an ITF involved in the CT), with the only difference being the ITF volume.

Under these circumstances, no experiment performed in any ITF can be considered the CT of any other experiment in a different ITF. In the reality, experiments recognized as CTs have been performed (list given in NEA/CSNI, 1996d, including the 2”-CL-SBLOCA in relation to which seven (7) experiments performed in five (5) differently scaled ITF are available, see e.g. D’Auria et al., 2005. Then, the following could be a more general CT definition, acceptable for the current S-SOAR, taking into account that any existing PWR (including VVER) can be the prototype for the ITF involved in the CT:

– The boundary and initial conditions (BIC) of CT experiments are properly scaled related to one prototype, i.e. according to a scaling method and a set of scaling laws accepted by each group of researchers managing an ITF involved in the CT activity.

To facilitate the comparability of experiments and phenomena, a CT campaign involving several ITFs may considerably profit from a preliminary agreement and, if necessary, an adjustment of BIC from a predefined scenario for each participating ITF. Any ITF experiment with a precisely defined BIC, even if already existing, may serve as a reference experiment.

In an attempt to minimize the effects of inevitable scaling distortions of the phenomena of interest, the following minimum set of BIC and parameters have to be ideally preserved between the potential CT experiments, e.g.:

– Same thermal-hydraulic state and parameters (pressure, temperature, mass inventory and flow condition) in each component of the facility;
– Same scaled values for power to volume ratio (Kv);
– Same scaled characteristics of primary and secondary-side safety and operational systems (e.g. accumulator injection and safety injection systems SIS characteristics);
– Preservation of the heat and mass sinks or sources (e.g. break location and size);
– Same timing of actions based on pre-defined operation criteria.

In practice, the conduction of CT campaigns usually leads to potential deviations from the above mentioned ideal configuration. This implies the compromises with respect to the specification of BIC. There exist quite a number of ITF experimental series being referred to historically as CT which do not match all of the above definitions and criteria. For such cases, the proper adaptation of BIC and a careful consideration and analysis of potential deviations to assure a valid comparison of key phenomena. That means a fundamental criterion for defining a CT, is to avoid, minimize, and quantify distortions in BIC between the experiments of consideration.

ITF experiments for common specific scenario, being selected for comparative purpose and whose BIC have not been aligned according to the above strict criteria of CT (due to being impossibility to pursue. or intentionally not being pursued) are referred to as “similar tests” (ST). For STs, facility-specific distortions of key phenomena, the timing of sequence of events or BIC, for instance, are accepted, and their quantification is not intended, and/or not possible. The definition for ST should be consistent with the CT definition. Therefore, the following statements apply:

- Any existing experiment in ITF can be involved in a ST activity.
- Overall, the transient scenario (e.g. SBLOCA, LBLOCA, LOFW, NC) and the expected phenomena (e.g. dry-out, maximum flow in NC) shall be qualitatively similar in different STs.
- Alignment of the BIC between the selected ITF experiments does not match the conditions for CTs proposed above.

However, both types of experiments above all significantly improve the understanding of a PWR thermal hydraulics, independent from computational thermal hydraulic analyses. While CT campaigns in particular place the focus on the investigation of the scalability of phenomena, the assessment of the codes' capabilities to scale phenomena and the development of scaling methodologies. The ST campaigns serve as basis for the identification and an in depth understanding of key-phenomena characteristic for the scenario, the specific test facilities, or their prototype system.

The value of a CT database should be seen as 'paving-the-way' to the process of extrapolating from the phenomena and the parameters measured in the different experiments to the NPP, i.e. one scaling bridge in Section 2.1.3.5. However, the key values for the resulting databases of both CT and ST are:

[code-to-experiment comparison] the possibility of demonstrating that the capabilities of system thermal-hydraulic codes are not affected by the scale, including the size of a facility;

i. [experiment-to-experiment comparison] demonstration of quality for the scaling parameters (including scaling methods and scaling laws) adopted for designing the concerned ITF and the experiment.

3.3.2 Merits and limitations in Counterpart Tests and Similar Tests

This Section describes a few examples of CT and ST and their impact on understanding the phenomena, focusing on the scaling aspect that may include both extrapolation and interpolation. The explanations are done first for the counterpart testing, followed by similarity testing.

<u>SB-LOCA Counterpart Tests in LOBI, SPES, BETHSY and LSTF</u>

One of the first, most complete and most widely used CT was the 6% cold leg break LOCA performed in the ITFs LOBI, SPES, BETHSY, and LSTF in the late eighties and early nineties of the last century.

All 4 test facilities simulate the primary circuit of a western type PWR, with original heights covering a broad range of volume-scaling factors: LOBI: 1:712; SPES: 1:427; BETHSY: 1:100; LSTF: 1:48.

The selected LOCA scenario is characterized by a complete loss of the HPSI, the early isolation of the steam generators, and a later injection from 4 cold-side accumulators at around 40 bars. In addition to the 4 experiments in LOBI, SPES, BETHSY, and LSTF starting at ''low" power (around 10% of the nominal power), two additional experiments in LOBI and SPES were conducted starting at high power (nominal power) to also cover the influence of an initially high power level on the relevant phenomena.

Relevant thermal hydraulic phenomena of interest during the CT campaign are the evolution and distribution of the RCS mass inventory, heat exchange with secondary side during degradation of the primary side with the reversion of heat flux, core heat-up and rewet in connection with loop seal behaviour, and the ACC performance.

As regards the overall development of the experimental transients, an initial core dry-out/rewetting induced by a depression in the level of the core liquid and clearing of the loop-seal was observed in all four ITFs.

The extension of this core dry-out can be correlated with the geometric difference in the main components (depths of cross-over legs). A second core dry-out due to boil-off before ACC injection was observed only in LOBI, and to a minor extent in BETHSY, and was counteracted by accumulator injection. A further core dry-out after emptying the accumulators occurred in all cases due to continuous inventory loss in the primary system that finally was stopped either by shutting off the core power (LSTF and BETHSY), or by initiating the LP injection system (LOBI and SPES).

A comparison of the overall experimental results, Annunziato et al., 1993, showed a good agreement with respect to the general course of events and the occurrence of similar phenomena in the individual tests. Differences in interactions, and in particular, the timing of various developments can be attributed to and explained by geometrical particularities, some specific design features, and boundary conditions.

The similarity of the overall results confirms the adopted scaling laws, and the suitability of the individual test facilities to reproduce a plant typical behaviour under the given boundary conditions. The general course of events is scale-independent, and can be expected to occur also in the PWR transient, assuming the same accident scenario. However, the time evolution of single parameters cannot be scaled-up or directly extrapolated to PWR conditions on the basis of the experimental database because the differences in results cannot directly be correlated with the overall (volume) scaling factor, but mainly result from differences in particular geometric configurations of the facilities and some operational differences in the tests.

The quantitative extrapolation and description of the plant behaviour are only possible on the basis of corresponding analytical analyses with T/H system codes. Extensive research has been carried out in aftermath of the experimental campaign dealing with the analysis of one or more of the above mentioned integral tests, or with the entire counterpart test campaign, and there are a large number of publications on this subject, e.g. Belsito et al., 1996, Ingegneri & Choinacki, 1997, and D'Auria et al., 1994a. Several system codes were used for post-test analysis of the experiments to evaluate the accuracy of T/H code calculations (by comparing the experimental results with the code results), to assess the capability of the codes to reproduce the observed phenomena at different scales, and to draw conclusions for simulating the plant behaviour and understanding it, with the final goal to verify that the code can predict the scenario expected in the reference plant.

The CT campaign on SB-LOCA was accomplished by an experiment in the PSB-VVER integral test facility in 2004. The PSB-VVER is a full-height ITF, replicating a VVER-1000 with power and volumes scaled at 1:300. From the analyses comparing the measured and calculated BIC from LOBI and PSB-VVER, similarities between the considered rigs can be seen, D'Auria et al., 2005. Furthermore, the main parameters (pressures, cladding temperature and RCS inventory) showed the same trends and similar thermal-hydraulic phenomena were measured during the tests. They demonstrate the similarity in behaviour of a PWR and a VVER 1000 for the case of a SBLOCA under scaled-down conditions. This fact confirms the validity of the CT methodology approach followed in addressing the scaling problem for this scenario.

In summary, it can be concluded that this kind of CT is well suited for assessing quantitative code capabilities and represents a valuable data source to improve the reliability of calculations with SYS TH codes.

PKL-2 and LSTF-2 Counterpart Tests

In 2011, counterpart tests were performed in the LSTF- and PKL-test facilities in the framework of the NEA ROSA-2, Nakamura et al., 2013, and NEA PKL-2, Umminger et al., 2013, projects (ROSA test 3 and PKL test G7.1), accompanied and followed by analytical activities that still are ongoing.

The selected scenario was an upward-oriented 1.5 % hot leg SB-LOCA, superimposed by additional system failures (no high-pressure safety injection, no automatic secondary-side cool down). A fast secondary side depressurization (SSD) initiated after the core was uncovered was employed as Accident

Management measure for restoring the secondary-side heat sink, aiming for a fast reduction in the primary pressure. The reduction of the primary pressure down to accumulator injection then effectuates the transition to the low-pressure phase with the low-pressure safety injection (LPSI) active. Besides investigating the efficiency of the applied Accident Management procedure, the main objectives of this counterpart test activity comprises questions on the following:

– Scaling effects between the PKL- and the LSTF-test facilities with special focus on
– Differences between core exit temperature (CET) and peak cladding temperature (PCT) under the same (but varying) boundary conditions in two differently scaled test facilities.

The issue on CET was studied in depth by the WGAMA Task Group on CET, NEA/CSNI, 2010a, who recommended a follow-up activity within the PKL- and ROSA-projects with pertinent experiments and/or analytical activities.

The initial- and boundary-conditions for the experiments in both test facilities were defined in close cooperation between the test facility operators (JAEA and AREVA) and the partners of both projects. Several code users undertook pre-test analyses and closely interacted with experimenters of both operators to help in designing a scenario to ensure compatibility with the facilities' construction requirements and to define the relative (volume/power) scaling factor between the two facilities.

In case the CET performance is involved, the most relevant part of the accident scenario starts after the primary side' pressure decreases to a value close to the pressure of the secondary side. The subsequent continuous loss of inventory on the primary side causes the core to heat up and CET to increase. This typically occurs under given boundary conditions in the pressure range from 60 to 80 bars. Due to this pressure limitation, this phase was simulated in PKL at a pressure level of about 45 bar, starting with already reduced primary-side inventory.

In ROSA/LSTF, two subsequent test runs were carried out, whereas the counterpart test run to PKL was performed with the same initial and boundary conditions as realized in the PKL test. Starting from a (quasi) steady-state at a secondary pressure set-point of 45 bar, the SSD was undertaken after observing a distinct heat-up (CET ≥ 350 °C) of the core for the shift to the low-pressure phase with ACC and LPSI (as it also was realized in the PKL test). To investigate the influence of the primary-side pressure on the CET performance, an additional, preceding test run starting from full pressure (160 bar) was conducted in ROSA/LSTF including the emptying of the primary side, and reproducing the occurrence of core heat up and CET performance in a pressure range between 80- and 50-bar.

The finally realized BIC (including opening the area of SG relief valves, injection characteristics, and set- point of the ACC- and LP-pumps) were in good agreement between the counterpart phases of both test facilities, allowing a direct comparison of the test results and the analysis of scaling effects.

The results of both tests showed a close agreement with the general trends of the main parameters, and with respect to main phenomena occurring during the transient. Both tests demonstrated the effectiveness of a secondary-side depressurization in the restoration / intensification of the heat removal from the primary side, and in the initiation of an accordingly fast (almost identical) reduction in pressure on the primary side. This effect is scale-independent, and the consequence of the intensive heat-transfer in the steam generators (condensation on the primary side, large heat transfer area). Due to the prototypical SG-tubing used in both test facilities, and the similar heat-transfer mode to be assumed under PWR plant conditions, the general trends/behaviour observed in the two facilities can be applied qualitatively to PWR plants.

One major objective of the CT campaign was related to the CET performance. In this respect, the different phases of the transient (reflux condensation, vapor superheating, and primary-side depressurization) reproduced under similar conditions (pressure, temperature, flow conditions) in differently scaled test facilities provide a broad range of experimental data for analysing the relation between the CET and PCT and the effectiveness of CET for AM actuations.

The comparison of test results showed a similar trend of the CET response (i.e. the ratio between CET and PCT) in the test facilities, and this trend can be applied qualitatively to PWR plants. However, because of unavoidable scaling distortions, and the diversity of influencing parameters, and the test facility specific design features affecting the CET performance (e.g. heat structures above the core in the vicinity of the CET position, the CET location itself, power profile, 3-D effects), the experimental results cannot directly be extrapolated in quantitative terms. Their interaction with code analyses is required, including a 3-D CFD analysis for this particular case wherein phenomena occur mostly due to the flow of superheated steam. Hereby, the experimental results from both test facilities serve as useful source of data to systematically analyse the effect of the particular design, and scale of the facilities on CET behaviour, to assess or improve the facility nodalizations and to draw conclusions for developing an adequate plant-nodalization.

In the meantime, intensive analytical studies addressing the above aspects have been undertaken by different organizations wherein analytical researches comprising post-test calculations and plant analyses in context with CET have been one major topic on the agenda of the joint PKL2-ROSA2 analytical workshop. Several ongoing efforts dealing with the definition of modelling guidelines for CET simulations with system codes, and with the development of new scaling-up methodologies based on the experiences from the LSTF-PKL counterpart tests were recently published, e.g. Martinez-Quiroga, 2014, see also Chapter 4 and Section 2.4.

In general, it can be concluded that the experiences gained from this counterpart test activity conducted within an international environment, have proven beneficial in understanding the fundamental phenomena and in initiating improvements in analytical techniques.

ATLAS and LSTF Counterpart Test

In 2015, a counterpart test was performed in the ATLAS facility to reproduce the LSTF test (SB-CL-32) in the framework of the NEA ATLAS Joint Project. The selected SB-CL-32 scenario was a 1% cold leg side-break LOCA, assuming the total failure of a high-pressure injection system and no inflow of non-condensable gas from accumulator (ACC) tanks of the emergency cooling system. Secondary-side depressurization of both steam generators (SGs) as an accident management (AM) action to achieve the depressurization rate of 200 K/h in the primary system was initiated 10 min after the break. Afterwards, auxiliary feed-water injection into the SG secondary-side was started with some delay.

Since the three-level scaling methodology was used when designing ATLAS, the scaling parameters for the counterpart test were inversely deduced by comparing the differences in geometry in both facilities; LSTF and ATLAS. Among several scaling parameters, primary inventory and core heated length were selected as reference parameters with first priority, from which other scaling parameters were determined, such as diameter, flow area, flow rate, and power.

The counterpart test results showed a slightly different loop-seal clearing behaviour because the reference power-plant of each facility has a different typicality in terms of the relative location of the active core region inside the reactor pressure vessel, as well as the depth of the loop seal. Nonetheless, overall very consistent thermal-hydraulic behaviour was evident from the counterpart test. Presently, analytical research is underway to address the scaling issue. It is expected that the uncertainties in extrapolating facility data to the real NPPs will be identified from this in-depth analysis in which many project participants are involved, by taking into account the geometric difference.

Counterpart Test on BWR

CT activities also were performed in relation to BWR ITF, as reported by Tasaka et al., 1985, and Koizumi et al., 1987, followed by Bovalini et al., 1992 and Bovalini et al., 1993a. The activities were respectively performed between ROSA-III and FIST first, and then among the ROSA-III, FIST, and PIPER-ONE facilities; both CTs on BWR recirculation pump-suction pipe SBLOCAs. The results showed some differences in the uncovering of the core and the heat-up behaviour, owing to different facility sizes, and the different scaling criteria adopted for designing the three facilities. However, discrepancies that arose among the different CT scenarios were 'expected' from the experiences gained on the facility response before the CT researches, and were explained as specification discrepancies (or differences between ideal and actually implemented boundary conditions), and thus were understandable. Merits and limitations of the CT activity were discussed in evaluating the computer code scaling capabilities (safety analyses tool, and for supporting the experimental CT activity as an extrapolation tool) and uncertainty analyses, Mascari et al., 2015.

Similar tests on Loss of Feed Water Transients

Similar experiments in several ITFs were conducted on the steam generator (SG) secondary-side bleed and feed, following a total loss of feed water (TLOF) accident (e.g. in LOBI, SPES, PKL, and LSTF).

This accident scenario is characterized by the complete boil-off of the SG secondary sides, associated with the loss of the secondary side heat sink that, in turn, leads to increases in temperature and pressure on the primary side up to the set-point of the pressurizer (PRZ) safety valves. Without any actions by the operator, the continuous loss of inventory via the PRZ valve(s) would result in the progressive uncover of the core. To prevent damage to the core, the secondary side bleed-and-feed is considered an effective emergency procedure in some plants to restore the secondary side heat-sink, so causing depressurization in the primary side due to condensation in the SG U-tubes.

It is expected that, due to flashing, a significant amount of water in the pressurizer is displaced into the RPV, sufficient to maintain or re-establish core cooling. In the experiments under consideration, the evolutions of the main parameters during the first part of a simulated accident scenario are rather consistent up to the initiation of the feed-water supply into the emptied SGs (taking place after the core heat-up in all the cited experiments). However, significant differences among the individual tests were observed with respect to the effect of the secondary-side feed on restoring the primary- to secondary-side heat transport and on core cooling; the latter is of main interest for evaluating the emergency procedure.

While in SPES, PKL, and LSTF the accident-management procedure employed was effective in restoring core cooling, the uncovery of the core could not be stopped in the LOBI experiment, which required further additional measures to limit a further increase in temperature. In LOBI, the injected feed water (injected in all ITFs about 10 m above the bottom of the SG) did not arrive at the tube sheet before the criterion for alternative measures (core temperature > 700 °C) was met. In other experiments, the arrival of feed water and subsequent onset of condensation on the primary side already was observed within about 100 s after the start of injection.

Extensive analyses including pre- and post-test calculations, parameter studies, and plant calculations were performed by different organizations, e.g. D'Auria, et al., 1992, Annunziato et al., 1992, Mazzantini et al., 1992, and Anoda et al., 1992, to explain the differences between the experiments, to evaluate the capabilities of system codes to correctly predict the main events and phenomena, and, definitely to draw conclusions applicable to NPP conditions.

The crucial point for the considered secondary-side accident management procedure (AMP) is the timespan needed by the injected water to pass the SG downcomer (DC) to reach the SG tube sheet, so inducing condensation on the primary side. This time span was significantly higher in LOBI (> 600 s) compared to the other test facilities (about 100 s or less).

Besides some differences in the SG operation and AFW conditions, the different hardware configurations of the steam generator DC were identified as the main reason for the different behaviour between the experiments.

- Annular DC in LOBI compared to other ITFs with external DC tubes,
- Narrow DC gap in LOBI.

In the LOBI experiment, the injected feed water evaporated completely at the superheated structures (hot wall phenomenon), or was partly held-up in the DC gap due to CCFL and did not induce notable condensation on the primary sides of the U-tubes before triggering the temperature limitation of the core simulator .

The CT campaign clearly demonstrated how different scaling approaches for important components can decisively affect the test results. So far, the importance of a 'correct'design or of the 'correct' interpretation of the results is evident. For all the ITFs, certain scaling distortions of the SG downcomer have to be taken into account (e.g. external DC tubes in SPES, PKL and LSTF versus narrow annular gap and oversized structure masses in LOBI).

In the computational analyses (pre- and post-test calculations) conducted at that time, this time delay could not be reproduced correctly. In contrast to the experiments, the codes identified the presence of feed water at the tube sheet and calculated the onset of primary-side condensation shortly after the start of AFW injection. Evidently, the codes underestimated heat storage in the structural masses, and heat transfer between the DC wall and the AFW, resulting in only partial evaporation of the AFW on its way through the DC.

Even though the observed time delay in LOBI was overestimated and is considered not representative of PWR behaviour under the given BIC (as indicated by subsequent PWR studies, Mazzantini et al., 1992, the analysis of this test and the comparison with the other experiments highlighted this scaling issue. In this respect, the analyses on the loss-of-feed-water experiments demonstrated the importance of correctly modelling the phenomenon of hot wall delay, including the application of a 'realistic' nodalization of the SG downcomer for the specific design of the PWR plant.

Similar Tests on Natural Circulation

As a fundamental mechanism of removing the core thermal power under normal and off-normal modes of operation, the performance of PWR natural circulation (NC) in the single- and two-phase conditions, including reflux condenser mode, is of high interest in reactor design, operation, and safety assessment.

NC results from the existence of a heat source (core) and a heat sink (SG), whereas the NC flow rate is established in a self-regulating way, according to a balance between the driving forces created by the density differences between the hot- and cold-branches connecting heat source and sink(s) and the counter drives (e.g. due to head losses from friction and singularities).

Fundamental experiments on NC behaviour have been carried out in all major ITFs, regardless of their reference NPPs (PWR, VVER, and CANDU) at least as part of the characterization testing during commissioning of the test facility.

Besides the systematic investigation of the transient behaviour (e.g. cool-down procedures without and with the isolation of SG on their secondary sides), parametric studies under quasi steady-state conditions with a stepwise reduction and an increase of the primary-side coolant inventory for different core power and pressure levels also have been conducted to systematically investigate the transition between flow regimes and heat-transfer mechanisms as function of the primary-side coolant inventory. Being characteristic for individual ITFs, the NC performance may be used also to address scaling issues. This performance measured in such experiments usually is demonstrated by illustrating the core mass' flow rate over the residual primary-side inventory. The ensemble of results from such similar tests (sometimes conducted as counterpart testing-campaigns) in different ITF provides an extensive database for the computational analysis of NC performance with T/H system codes.

In spite of partly considerable differences between the individual experiments with respect to the test facility design or test procedures, a similar, general trend identifying five different flow-regimes dependent on the residual inventory can be derived from the results for the PWR:

– Single-phase NC with no steam in the primary circuit,

- Stable co-current two-phase NC with inducing increased loop-flow rates,
- Unstable two-phase NC with pronounced flow oscillations (after passing the two-phase flow maximum), e.g. fill and dump (…heterogeneous behaviour…),
- Stable reflux-condensation (RC) with condensed liquid flowing back to the core via the hot legs and cross-over legs,
- Core dry-out situation.

The power and pressure dependent transition between the different flow states may be illustrated in Natural Circulation Flow Maps (NCFM). The NCFM allow a direct comparison between experimental results acquired from ITF, e.g. D'Auria et al., 1991.

In 2014, a new natural-circulation database at high pressure and under different power conditions was added to the existing database, utilizing the reduced-height facility, ATLAS. Such tests were considered to be valuable to support the NCFM, as the NC flow rate was measured at the reduced-height facility rather than at a full-height facility.

The same test procedure was undertaken as for the previous NC tests. The primary inventory was drained stepwise during the test, and the natural circulation regime passed from single phase NC with no steam through unstable two-phase NC to the core dry-out situation.

It turned out that the measured non-dimensional NC flow rates were within the envelope of the existing full-height database.

In the proposed NCFM, two-dimensional parameters were used: (1) Mass flux divided by power; and (2) the remaining inventory. However, it is questionable whether the two-dimensional parameters are appropriate in representing the NC flow. In particular, in the present domain, the initial power is greatly influencing, implying that a small core power results in a large variation in the x-axis. If we take into account the heat loss in the entire NC-flow system, it results in much uncertainty in the NCFM. Thus, it is worthwhile to revisit the two parameters to ascertain whether those parameters are the best ones governing NC flow in a loop.

Comparative analyses on the scaling of different flow phenomena were pursued on basis of data from the ITF characterization tests, D'Auria et al., 1991, D'Auria & Frogheri, 2002, and Cherubini et al., 2008, or by dedicated bi- and multi-lateral CT campaigns, e.g. PKL versus LOBI, Kirmse, 1992, BETHSY versus LSTF, Bazin et al., 1992. The considered experiments from the different ITFs all showed qualitatively similar behaviour: A maximum core- and loop- flow- rate in the two-phase regime, and a decrease to almost zero loop flow under RC conditions. While the general trend can be applied to a NPP scale, a direct scale-up in quantitative terms from test facilities to NPP is not possible, due to differences in the ITF design and operating modalities (e.g. specific geometric details or operational aspects, such as bypass flows in the RPV or PRZ configuration), despite the rather large spectrum of volume-scaling factors covered (e.g. Semiscale to LSTF). This is of an even more vital importance for scaling of the two-phase flow regime, a process which proved to be challenging even with current analytical tools.

In an attempt to formulate rational criteria for the quantitative extrapolation of the NCFM data measured in small scale facilities, D'Auria et al., 1991, confirmed that the utilization of qualified computer codes was the appropriate approach.

In respect thereof, ITF parameter studies on NC behaviour, in particular if collected for different powers, mass inventories, numbers of loops, numbers of SG U-tubes, facility heights and pressures from ITFs participating constitute a valuable database with clear BIC for validating T/H codes against the transition between different flow regimes. If a code can correctly predict the change of flow regimes for different ITFs from different scales, a reliable prediction of these fundamental mechanisms for NC behaviour in PWR may be assumed.

For predicting the occurrence of the different flow regimes during PWR accident transients, a comprehensive data-base comprising experimental results on all five flow regimes, along with particular characteristic separate effects (e.g. the CCFL during the RC condition) from different ITF are available to clarify all the fundamental mechanisms for NC behaviour in a PWR.

Difficulties in simulating separate effects associated with the simultaneous heterogeneous behaviour of individual U-tubes (e.g. fill and dump, stagnant, reverse flow) resulting from the lack of multi-dimensional calculation models for the SG out- and inlet plenums still challenge the quantitative prediction of transients involving such effects by means of nowadays T/H system codes. This may limit the quality of calculated results for the reference reactor.

<u>Conclusive remarks on CT and ST</u>

The prediction of the plant behaviour under postulated accident situations in qualitative- and quantitative-terms within acceptable uncertainty limits is the major goal of safety analyses. CT and ST can significantly contribute to the solution of this task. The benefits of CT and ST can be summarized under the following three aspects:

– Demonstration of phenomena expected to occur in the NPP under the same accident conditions.
    - o The occurrence of certain phenomena in differently scaled facilities confirm, or at least remarkable increase, the confidence that these phenomena will also occur in the reference NPP. In some cases the results or findings from differently scaled facilities can be applied to reactor conditions without additional code analyses.

– Demonstration of the suitability of the individual test rigs and the tests to "adequately" reproduce the reactor typical behaviour.
    - o CT constitutes an effective way to analyse/understand scaling effects, and the influence of scaling factors, scaling distortions or certain particularities of the facility on specific phenomena, or on the overall system performance.

– Database for code validation
    - o CT provides an important contribution to qualify the overall code validation process, and to improve concepts of code simulation and the predictive accuracy of code applications for full size NPPs. In particular CT, following Karwat, 1985, and Karwat, 1986:
        - Helps to identify and characterize errors or uncertainties of code analyses;
        - Supports quantifying scaling analyses;
        - Assesses up- and down-scaling capabilities of codes.

However, CT testing in ITFs may not always suffice to achieve a reliable quantification of the code uncertainties in predicting certain relevant phenomena. In such cases complementary information for assessing and improving the code model must be the result of independent confirmatory analyses performed using data from separate effects tests, Glaeser & Karwat, 1993.

3.3.3 Experiences from Daughter Facilities

For some scenarios, certain relevant phenomena dominated by highly-heterogeneous, three-dimensional flow patterns cannot be reproduced by sub-scaled ITFs due to scaling distortions imposed by constructive compromises. For such applications, complementary tests in test facilities dedicated to investigating separate effects, that is, SETFs become invaluable.

Experimental set-ups in separate effect test facilities offer significant advantages, such as a clear set of boundary conditions, the possibility of adjusting or focusing the instrumentation on particular phenomena, or even employing whole test rigs dedicated to specific phenomena. There also is the possibility of more systematically evaluating the accuracy of calculation models over a wide range of conditions under steady-state- or transient-operations (parameter studies). In the frame of complementary testing, whereas the ITF concentrates on studying the overall system response, the SETs investigate the responses of the plant subsystems, and, in particular, study individual phenomena that are highly dependent on the geometry, in scales up to 1:1 full-scale (in the case of the UPTF).

This category of tests/set of tests gathers ITFs and SETFs in a literally complementary way to capture all relevant aspects of an accident transient so to provide a realistic picture of a scenario that often encompasses multiple safety aspects. The integral system behaviour, significant coherences, and relevant boundary conditions are illustrated by ITFs, whereas separate important phenomena may be recorded in a significantly higher level of detail (e.g. 3-D-effects) by suitable SETFs in the frame of complementing experiments. In this way, not only one-dimensional BE computer codes benefit from testing campaigns, but also 3-D-modules and/or 3-D-modules to be implemented in T/H system codes for adequate components may benefit from acquiring data at high resolution in space.

Another category of tests or combination of tests referred to as daughter (facility) tests employ results available in 1:1 full-scale as reference for comparison with results from scaled-down experiments on the same phenomena, and aims at evaluating the scalability of relevant phenomena and their understanding in general. In this way, the representativeness or the limits of scaled down facilities with respect to certain important phenomena can be analysed, and the impact of scaling distortions on the overall system behaviour during accident transients evaluated.

In the following, some typical examples for both categories of complementary tests and daughter (facility) tests are described:

– CCFL in hot legs (daughter-facility test),
– Coolant mixing in DC (complementary testing).

<u>Daughter Facility Tests</u>

CCFL in the Hot Legs

At a significantly reduced inventory of primary coolant water, the reflux condenser mode of natural circulation may occur as described in previous Sections. At very high levels of heat transport in the SGs, condensate backflow from the SG U-tubes to the core via the hot legs can be impeded, as a result of high steam velocities in the opposite direction. This effect, known as counter current flow limitation (CCFL), which is expected to occur under certain boundary conditions in the hot legs, the inlet of the SG header and in the SG U-tubes, plays an important role in many accident sequences because it controls a possible re-distribution of inventory from the core into the direction of the SG, i.e. the amount of coolant that is kept outside the core, and is no longer available for cooling the core. The occurrence of CCFL directly is connected with the prevailing velocity of the steam, and mainly is influenced by the primary pressure and the decay power, or more precisely, by heat transport to the SGs. Low pressure levels (low steam density) and high power levels (high steam mass-flow) promote the occurrence of CCFL, whereas the transport of SG heat is determined not only by the core decay power but also by the procedures applied. Cool-down gradients on the secondary side, and the subsequent decrease in the primary side pressure, and the secondary- side isolation of one or more SG, for instance, result in higher steam flows towards the active SG. In addition, the occurrence of CCFL is highly dependent on the geometrical configuration of the relevant components (HL diameter, bend to the SG inlet header, SG inlet header, and inlet to the SG U-tubes).

Counter current flow in a PWR hot legs was investigated at different sub-scale SET facilities with pipe diameters up to 200 mm (see Section 4.3.4.4 and Figure 4-17). To provide CCFL data for full-size geometry, systematic investigations on CCFL in the hot legs at different levels of pressure (3 bars, 15 bars) were also performed in the 1:1 scaled UPTF test-facility. The main results in comparison with the sub-scaled facilities, also show corresponding issues related to scaling distortions; the consequences for code modelling are discussed in chapter 4.

However, CCFL on the hot side (hot legs up to the SG U-tubes) also is an important aspect in many reduced-scale IETs characterized by a significant reduction in inventory and high SG load. On the one hand, CCFL has been systematically investigated within parametric studies under quasi steady-state conditions with variations of the relevant parameters (pressure, decay power) in several IETs. In addition,

CCFL in the hot legs also was observed in many transient tests covering a broad spectrum of accident scenarios (e.g. SB-LOCA, SBO with secondary-side AM measures, partly under asymmetric conditions). The occurrence or absence of CCFL also is an important factor in certain cases for evaluating the effectiveness of countermeasures. ‘Too intense’ heat transfer to the secondary side resulting from a very fast cool-down of the secondary side, or from injection of large amounts of cold water into emptied SG secondary sides can lead to an unfavorable displacement of water from the core to the hot legs and the SG inlet chamber/U-tubes.

Therefore, it is therefore important to know the CCFL characteristics of the IETs (also for the hot-leg configuration), and to what extent the CCFL behaviour in the hot legs in the individual IET facilities is representative for the PWR, and to evaluate a possible scaling effect on the overall system behaviour (e.g. core cooling). The results of the UPTF tests results, the comparison with smaller-scale test facilities clearly showed the dependency of the geometry on the CCFL behaviour in the hot legs: The discussion in Chapter 4 demonstrates the difficulties in appropriately scaling the complex geometry of the hot leg and the region up to the SG U-tubes, i.e. the bend and the inlet chamber in scaled-down IETs. In this respect, the results from the full-scale UPTF on CCFL in the hot legs can serve as reference for corresponding tests in scaled ITFs (in this sense, considered as a daughter test-facility) and can be used as orientation when defining the boundary conditions, and interpreting the results of IET tests, or, as far as possible, in adapting the geometry of the hot leg accordingly.

As a consequence of the findings from UPTF tests on CCFL, it was decided to re-design the hot legs in the PKL test facility (deviating from the ‘pure’ conservation of the Froude number, the diameter was enlarged, and the SG inlet header and transition between the horizontal part and SG inlet header were modified) to assure the occurrence of CCFL in the hot legs at the same (scaled) heat transfer to the SG as in UPTF (for the same pressure levels). The objective of this measure was not to investigate CCFL in the hot legs of the PKL, but to avoid the impact of the reactor atypical behaviour in the hot legs on the behaviour of the overall system. In particular, ‘clean boundary conditions’ for investigating CCFL in the SG U-tubes could be realized, i.e. it could be assured that the accumulation of water in the U-tubes is due to CCFL in this region, and is not the result of growing swell levels in the SG inlet header up to the U-tubes, initiated by CCFL in the HL.

The tests in the UPTF- and PKL-test facilities on CCFL in total provide a good picture on the CCFL behaviour in the region from the hot legs to the SG U-tubes, at least for plant configurations like, or similar to German Konvoi plants, Umminger et al., 1997. The test results have shown that during SB-LOCA under design-basis accident conditions, CCFL has not to be expected, neither in the hot legs nor in the U-tubes during cool down with 4 SG above a primary pressure of 10 bars. However, under certain adverse conditions, CCFL and the corresponding water re-distribution out of the core have to be taken into account.

<u>Complementary Tests</u>

Coolant mixing in RPV Downcomer

Under certain accidental conditions (e.g. breaks in the main steam-line breaks, MSLB, and SB-LOCA) coolant entering the reactor pressure vessel of a PWR via the inlet nozzles may have a different temperature and boron concentration different from those present in downcomer. In absence of proper mixing, differences in temperature may lead to unacceptable thermal gradients, affecting the structural integrity of reactor pressure vessel (RPV) due to pressurized thermal shock (PTS).

For the SB-LOCA case, the possibility of coolant with a relatively low boron concentration collecting in localized areas of the RCS has been discussed for several years. Causes might be the injection of coolant with less boron content from those in interfacing systems (external dilution), or the separation of the borated reactor coolant into highly concentrated and diluted fractions (inherent dilution). Examples of external dilution are the injection of coolant with a lower boron concentration by the makeup system, and injection of low-boron pump-sealing water into the primary system. Inherent dilution can occur after the transfer of reflux-condenser heat or a backflow from the secondary system in cases of primary-to-secondary leakage accidents. Operation in the reflux -condenser mode over a lengthy period could occur in the event of SBLOCA concurrent with limited operability of the emergency core cooling (ECC) systems. In such events the condensate descending down the tubing of the cold-side of SG U-tubes into the SG outlet plenum, and from there into the pump seal, could form slugs of low-boron water. On restoration of natural circulation after refilling of the RCS, such slugs would be transported towards the reactor core.

During the non-isolatable MSLB, a rapid decrease in secondary-side pressure in the affected SG increases heat transfer from the primary- to the secondary-side, and therefore, to pronounced cooling of the primary coolant in the affected loop (sub-cooling transient). An important question during this process is whether a localized re-criticality of the core, and the resulting power excursion, can occur due to the entry of cold water into the area of the reactor core. Furthermore, an additional, important aspect of this accident scenario concerns the RPV integrity under the consideration of PTS due to the discharge of cold water in the RPV downcomer. This is important above all when the cooling of the primary coolant is intensified by the injection of emergency cooling-water into the cold leg at high primary-side pressure (up to the actuation pressure of the pressurizer safety valve).

For both exemplary cases, the coolant on its way to the core would be mixed in the cold-leg piping, the RPV downcomer, and the lower plenum. The investigation of the hypothetical re-criticality or PTS events is a topical issue in PWR safety-analyses for both SB-LOCA and MSLB scenarios. A deeper understanding of the physical processes connected with the coolant mixing is needed to assess realistically the possible consequences of such events for the reactor core, or the RPV structures, respectively. In recent years, renewed focus was placed on studying these scenarios, with emphasis on the analytical description of the mixing phenomena, for assuring a deep assessment of the phenomenology in the cases of core reactivity increase and PTS on the RPV walls.

For both cases, the main mixing mechanism – assuming safety injection (SI) being mostly unavailable in the case of SB-LOCA – occurs in the RPV in the RPV downcomer, and is buoyancy-driven turbulent mixing. The density differences between the fluids are due to the differences in temperature and possibly boron concentration. The computational tools that are needed to analyse the mixing of coolant flows on their way from the loop to the core inlet have to be validated by suitable experiments.

Extensive experimental- and theoretical-studies have been conducted in this area; 3-D-mixing effects in the downcomer imposed by density differences between the different coolants in the facility (and reactor) were identified as playing an important role for the safety-relevant phenomena in these scenarios. Where geometrical configuration imposed by the constructive compromises necessary in ITFs limit the transferability of measured data, the relevant issues (PTS at inlet nozzles, or mixing phenomena in the downcomer) are addressed by complementary experimental tests in suitable separate-effect test facilities.

In connection with boron-dilution events, complementary tests in the test facilities PKL and UPTF were performed in the nineties. While the PKL system test facility was used to study the start-up of natural circulation following the refilling of the primary system after reflux condensation, the 1.1 scaled UPTF was used for tests on mixing of flows of water flows with different boron concentrations in the cold legs, and the RPV downcomer annulus, Hertlein et al., 2003.

Owing to its dimensions and design (full-scale in terms of height, and symmetrical layout of the four loops), the PKL test facility is well suited for studying natural-circulation phenomena. However, due to the required constructive compromises, the PKL test facility is not designed for a realistic simulation of mixing processes, in particular in the downcomer annulus, and in the lower plenum. This topic was investigated in the context of the tests conducted in UPTF. As a result of the full-scale simulation of the RPV, including the cold legs and the ECC injection nozzles, as well as the downcomer and the connected primary system lines, the UPTF was ideally suited for investigating these situations.

Later, after the dismantling of the UPTF, the ROCOM test facility, Kliem et al., 2008, and Kliem et al., 2007, was employed as a further source of complementary data for PKL system tests for both accident-scenarios (SB-LOCA, MSLB) so to complete the spectrum of possible flow conditions for mixing, and to allow for an even more detailed analysis of mixing phenomena in the downcomer annulus and the lower plenum. Within several complementary test campaigns (performed within international OECD projects), the results from PKL integral system tests have been used to define the boundary conditions (e.g. temperature and natural circulation mass flow rates at the inlet to the RPV for the individual loops).

The combination of the PKL and of the ROCOM experiments on MSLB and SB-LOCA covered the main thermal hydraulic phenomena relevant for these scenarios. The test results were used extensively y the project partners for validating and optimizing analytical tools, which are for system codes in connection with PKL, and for CFD in connection with ROCOM. Apart from validating the CFD codes for replicating the relevant phenomena, the CFD-qualification grade data obtained in this way by the complementary experiments is used to support the further development and validation of 3-D modules to be implemented in the BE system codes for the relevant RCS components.

In summary, the abovementioned examples illustrate the significant contribution provided by the close collaboration between ITFs of different scales and SETs in determining the scalability of phenomena, the validation of computer codes, and the understanding of complex thermal-hydraulic phenomena associated with nuclear safety issues in general. Although a quantitative determination of relevant phenomena for PWR-scale requires the employment of BE system codes, complementary testing may provide an invaluable contribution for estimating the uncertainties involved in computer calculations.

3.4 Scaling roadmap for designing a test facility

The H2TS, FSA, and DSS methods are hierarchical scaling methods that can be used to design a scaled integral-system-test facility to examine a predefined set of phenomena. In some studies, this predefined set is obtained using the PIRT process, or by performing an order-of-magnitude assessment of the individual effect parameters in the balance equation. The DSS method recognizes that the relative importance of an effect parameter may change as the transient evolves. Therefore, the DSS examines scale distortions over the entire duration of the transient. All three methods include the scaling of local phenomena.

Figure 3-7 shows an example of scaling roadmap, in the form of a diagram for applying DSS to hierarchical systems, so to obtain integral test-facility design specifications. A box labeled Local Phenomena is highlighted in yellow which requires specific models and correlations on local phenomena as input into effect parameters that are used to scale the system or components. In H2TS, this is known as "bottom-up" scaling. This approach was used to scale all the important local phenomena in the APEX-600, APEX-1000, MASLWR, GRTS, and NIST-1 integral test facilities Reyes et al., 1995, Reyes & Hochreiter, 1998, Reyes, 2004, Welter et al., 2005, Reyes et al., 2007, Reyes et al., 2010, and Reyes, 2015a.

This scaling roadmap includes six self-explanatory main steps, [1] to [6], in Fig. 3-7.

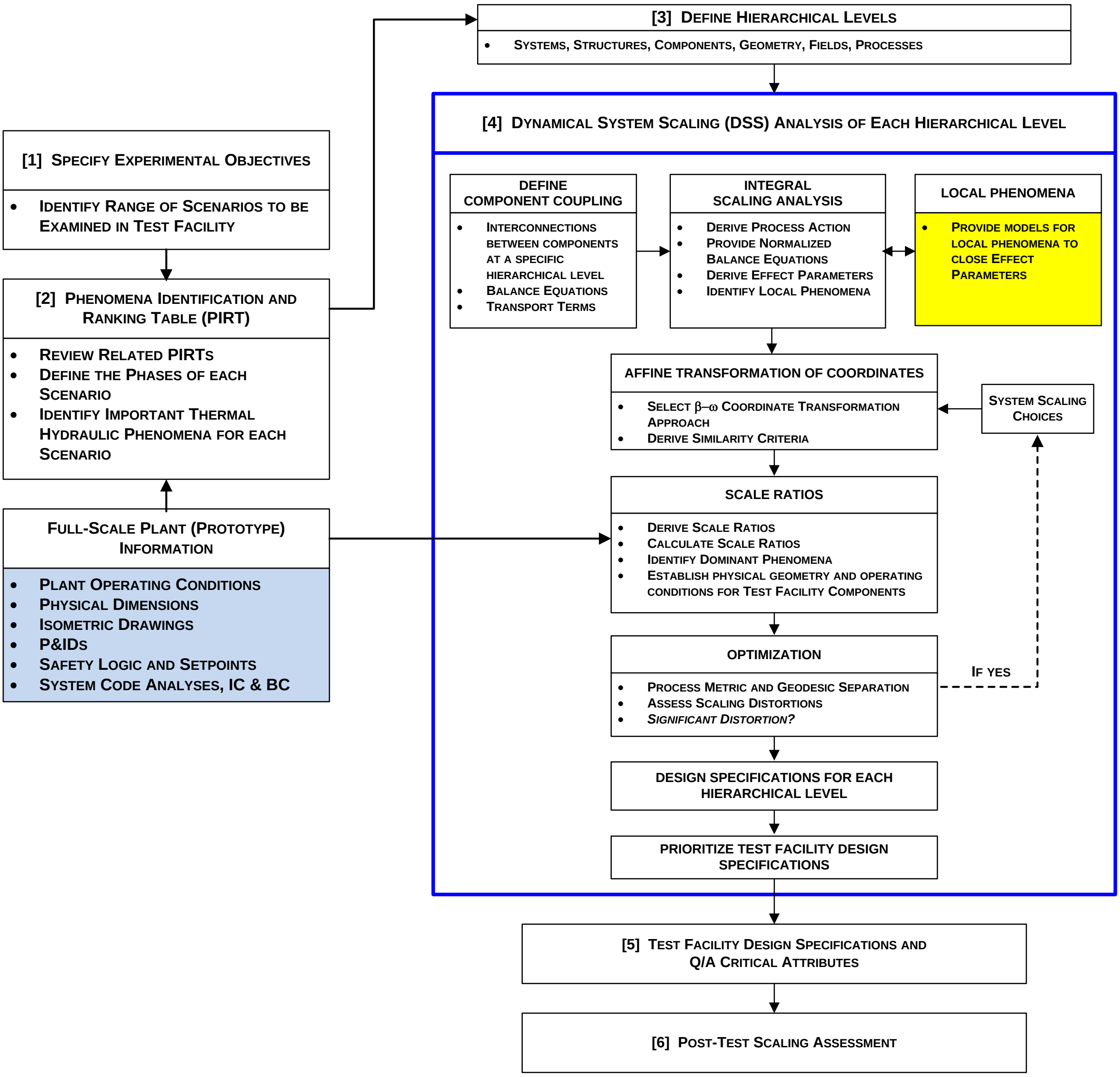


Fig. 3-7 – Scaling roadmap adopted for the design of test facilities by DSS

The concerned facilities provided benchmark data for Westinghouse, INL, and NRC T/H systems codes for the primary loop natural circulation, LBLOCA, SBLOCA, MSLB, PTS overcooling scenarios, and Station Blackout, using active or passive-safety systems. The APEX-CE test also had a highly instrumented downcomer, so it provided data on the thermal fluid mixing in falling plumes to benchmark CFD codes. The following phenomena were considered in the local phenomena scaling for the design of the facilities:

– Loop and individual component loss coefficients, including preserving scaled loss coefficients for the core, core-flow bypass and entrainment in upper internal structures. Typically prototypic spacer grids and fuel-bundle pressure drop tests are conducted in separate effects tests. Similarly, CHF tests are undertaken in separate effects tests.
– Thermal fluid mixing plumes in the downcomer and lower plenum. This can be applicable to boron

mixing.
- Onset of counter-current flow and diffusion in air and helium gases (e.g. CCFL in the presence of incondensable gases).
- Transient heat conduction across gas cooled reactor core
- Transient heat conduction through a containment wall with steam condensation and free convection heat transfer boundary conditions.
- Onset of thermal fluid mixing in the cold leg due to counter-current -fluid flow caused by high pressure injection.
- Steam-liquid CCFL in a pressurizer perforated plate (i.e. the electrical-heater support plate).
- System heat losses.
- Breaks and valve leakage under choked flow and non-choked flow conditions.
- Reactor pressure vessel depressurization and containment pressurization.
- Onset of siphon-condensing in two-phase natural circulation.
- Onset of flow regime transitions.
- Many more local phenomena.

3.5 Key findings

Scaling techniques and relevant experiment efforts are reviewed in Chapter 3.* This overview comprises the description of design approaches required for facility design, major scaling methods used to address the scaling of local phenomena, and the system integral response of reactor system via the scaled test facilities for separate-effect and integral-effect (SETs and IETs) for various reactor types such as PWR, BWR, VVER, and new LWRs, as well as the containment vessel. Counterpart testing and daughter facilities are discussed in the attempt to consider the influences of scaling

Findings from the overview are summarized as follows;

1. A scaling approach is an essential step to provide the “best” scaled facility design to generate experimental data for developing models, correlations, and closure laws as components of computer codes, and to validate the computer codes. A scaling approach is also needed to identify and characterize the hierarchy of scaling factors affecting thermal-hydraulic phenomena at each level of local, component and reactor system. In addition, scaling methods are also used to estimate scaling distortions. In the case of facility design, scaling analyses are performed taking into account available resources and distortions are inevitable.

2. Scaling method(s) are selected in conjunction with the scaling approach to design a “best” scaled test-facility and to understand the data from the test facility SET and IET. Appropriate scaling factors are defined corresponding to the thermal-hydraulic phenomena at each level, local, component, and reactor system, which may appear during experiments to simulate prototype in the scaled facility.

3. Non-dimensional key groups to describe local phenomena that may involve whole reactor transient are obtained, and used to design test facilities. These groups include, both traditional ones such as Re, Fr, Bo and Pr and also newly-created ones such as those adopted to fulfill the three-Level scaling method proposed by Ishii. In any case, an applicable and thus valid range may exist in the thermal-hydraulic conditions (such as geometry, fluid property, mass flux, heat flux, phase change, multi-dimensionality) for each of non-dimensional group, according to the range of conditions for the experiment from which the non-dimensional groups were developed.

* Role and effectiveness of experiments on the validation of local phenomena models/correlations and computer codes for system-wide response are discussed in Chapter 4, as well as the role of scoping analyses using system analysis code(s). These descriptions are made with appropriate reference to the related portions in Chapter 3.

4. Related to item 1 above, scaling distortions should appear in any scaled test facilities and, thus, in their data because all the relevant scaling factors are not ideally matched in the facility design. Therefore, quantitative (direct) extrapolation of the experimental data to the prototype is not feasible. Results from the scaled (simulated) test facility are not expected to represent the replica of prototype phenomena.

5. Basic information on most of SETs and IETs, including design features relevant to scaling, which are suitable for model development and code validation are collected in Chapter 3 and in Appendix A3. They are classified considering reactor designs of PWR, BWR and VVER. For both SET and IET, many of boundary conditions and key scaling parameters that should be considered in their design stage are described as derived from published literature. Many examples from notable test facilities are used to obtain important findings concerning the accident phenomena. Key scaling parameters include time, height, length, volume, fluid, pressure and major reactor components. One of the main outcomes is that the data obtained from an ITF cannot be directly applied to full-scale prototype. As for the SETFs, the initial and boundary conditions are very important in considering the validity and applicability of the obtained data, even when the geometry is a replica of (or a part of) a component of interest of a reference reactor. Much effort has been applied to facility design and operation to minimize scaling distortions in the results.

6. Scaling considerations for the containment vessel and the advanced reactor design are done separately from those for conventional reactors (BWR, PWR and VVER, discussed above). Simulations were scarce for the containment ITF in the earlier experiments, partly because the actual size containment of a reactor (though having smaller size than typical commercial reactors), such as an HDR, was employed after decommissioning the reactor. As the phenomena in the containment are clarified through experiments, phenomena scaling came to be one of the key subjects to be considered. A few key phenomena, such as the simultaneous interaction of a large number of vent pipes in BWR Mark-II containment, indicate that essentially different type of phenomena need to be addressed in CV.

7. Connected with item 4, above, scaling distortions, inevitable in all the experiments, make it difficult to directly extrapolate the data to prototype (or apply those data to prototype). When the similitude of a facility is sharpened by concentrating on certain phenomena of interest, a greater degree of distortion may happen in other phenomena in the scaled experiment. Major scaling distortions, such as the flow in a pipe, structural heat storage, pressure drop, multi-D phenomena, pump, and fuel are pointed out with possible explanations.

8. While our test facilities are full of scaling distortions, an attempt has been made to clarify such scaling distortions, and the possibility of extrapolation/interpolation of the obtained data. Counterpart tests and similar tests then have to be used for this purpose; they provided a great amount of information to promote the understanding of accident phenomena observed mainly in ITFs at different scales. This understanding enhances confidence that these phenomena may also occur in the reference NPP, and is suitable for the computer code assessment. However, further extrapolation of the results beyond the scale of the test facility with the maximum scaling ratio (nearest to the prototype) is not guaranteed, so providing an upper limit of applicability of the knowledge from the counterpart testing, as long as the maximum sized facility also has scaling distortions.

9. Daughter and complementary tests constitute “previously well-planned” sets of experiments to clarify the influence of scaling in the observed phenomena in differently scaled test facilities, respectively addressing the reactor system response (ITF versus ITF) and the local phenomena (ITF versus SETFs). The idea of former sets of experiments is close to that of the counterpart tests. These are other types of experiments to provide information about scalability into the experimental results. A database useful to validate the code predictive capability is obtained; this is also suitable for increasing confidence in the reactor safety analysis by using the validated code. Especially, complementary testing may provide a valuable contribution to estimating the uncertainties in code predictions.

4. SCALING AND THE SYS TH CODE

4.0 Introduction

Scaling methods, such as H2TS and FSA, have proven to be very useful tools in analysing a complex problem, to identify dominant processes, to support the PIRT analysis, to scale adequate SETs and IETs, and to synthesize the data obtained in a useful manner for applications to reactors. Scaling methods are used to design facilities that minimize distortions in important phenomena over the range of the transient of interest. Their promoters, e.g. Zuber et al., 1998, considered that these methods could reduce the use of system codes in demonstrating safety. Transients are divided into phases based on their complexity and the change of dominant phenomena. SBLOCA is divided into five phases, while LBLOCA has three phases only. The system can be divided in space and time to account for complexities.

However, system codes exist which also have integrated the knowledge gathered from the huge data base produced so far for LWRs, and which can help at every step of the analysis of complex transients. The cost of CPUs no more is problem for system codes, and manpower cost now is higher than that for CPU by orders-of-magnitude. System codes can reduce the cost of manpower by quickly doing evaluations that are done "by hand" in H2TS and FSA. System code is not an alternative to scaling analysis, but is a tool for assisting scaling analysis and solving problems. For example, the identification of the phases of a transient may be easier after some preliminary simulations of the transient using system code, even before having simulated the transient with an appropriate IET.

H2TS and FSA identify the dominant processes at the system level starting from balance equations, writing them in a non-dimensional form, and evaluating each term for dominant phenomena/ components represented by terms in the global balance. This facilitates a reduction in the search for process models, which are important contributors to selected figures of merit, to fewer components and objects for minimizing scale distortion. System codes also use balance equations and constitutive relationships. They can identify the dominant processes. They also more easily can predict how the relative importance of each process may change all along the transient. The evaluation of the relative importance of each process can only be a very rough approximation in scaling methods and codes may be much more precise, provided that they are qualified with V & V.

System codes open the possibility of investigating phenomena that may be of second order importance, but might require some attention. Many phenomena may not be recognized as dominant by scaling methods, but system codes may reveal that they have a non-negligible quantitative effect on the Figure of Merit. For example, a bypass flow from downcomer to an upper head may have an effect on the loop-seal's clearing process in a PWR SBLOCA, or during the reflooding in a LBLOCA, and no IET may be fully representative of all PWR designs with respect to this bypass. Parametric studies using system codes can investigate the effect of this bypass in a few hours using a small computer, whereas parametric studies using an IET would need months of manpower. This will require appropriate modelling in the codes. Then system codes easily can help the analyst when identifying the relative impact of various processes.

In a complex system, the interacting components may create phenomena that are difficult to include in a preliminary PIRT, and in the scaling analysis, simply because they are difficult to understand. In particular, instabilities may occur which depend on the interactions between the components of the reactor circuit; the transient performances of a complex system may be analysed by system codes and that help to clarify them. As an example of mutual influence between global- and local- phenomena, we can consider the depressurization rate, global phenomenon, and the critical flow at a break, viz., local phenomenon. This interaction here is easy to identify in the PIRT. A more complex interaction exists during the beginning of a reflooding phase in a LBLOCA. Strong oscillations exist between the core and the downcomer. The local behaviour of droplets entrained from the core to the steam generator during oscillations with partial deposition, and their possible re-entrainment and vaporization in steam generator after some transit time, may affect significantly the damping or amplification of the oscillation, and the possible loss of ECCS water to the break. As a consequence of these local processes in the hot leg and the SG inlet plenum, a significant impact on the second- and third-peak clad temperature may be observed. This complex interaction may not have been identified as a dominant process by the preliminary PIRT, and both the analysis of IETs and the use of the system code's sensitivity tests can contribute to gaining a more precise PIRT.

System codes can check the adequacy of the ranking of processes by calculating a transient performed in several IETs having different scaling ratios, and by investigating whether extrapolating the scale to the reactor proves or disproves the ranking.

System codes also can study the effects of distortions in IETs, and may provide more reliable extrapolations to the case of a reactor. They may prove their capability in predicting the distortion using appropriate SETs that investigate the process of interest with and without the distortion. In this way, system codes can quantify the scale distortion due to some processes that could not be properly scaled in the IET design.

The up-scaling capabilities of a system code depend mainly on how it can predict phenomena that are distorted in scaled IETs. It also requires that all important physical processes that may play a significant role in the transient are modeled correctly.

Even if system codes have many capabilities, they also introduce some distortions to the reality, due to simplifications of the physics, to non-modeled phenomena, and to the limited accuracy of the closure laws. Thus, there are many requirements for a reliable system code application to safety. This includes requirements for selecting the model, for developing the code, for its validation and verification, and for its proper application, and scaling considerations are present during all these steps.

The code must be able to predict global parameters in any IET, such as system pressure, and mass inventory, and to predict with the same reliability the same transient at different scales using counterpart tests; the code also must predict important local parameters, such as the clad temperature in SETs and IETs. It also should be able to predict correctly at the reactor scale those phenomena which are distorted in IETs

The current generation of best-estimate system thermal-hydraulic codes were used in previous PIRT analyses of several accidental transients, and in scaling analyses for selecting the adequate type of modelling, the acceptable simplifications, and then, for defining the requirements of validation and verification. For example, a 1-D calculation of heat conduction in solid structures is used in most cases, but a 2-D-conduction calculation was implemented in several codes to predict the progression of the quench front during core reflooding. This was the result of the identification of the role of axial conduction in the process, and of the necessity to use the code to extrapolate from re-flooding experiments to the reactor because many experiments are not well scaled with respect to the effects of gap conductance, the fuel pellets' conductivity and heat capacity.

In this Chapter, the code's merits and limits with respect to scaling first are presented below, considering the processes of code development, code verification, and code validation. The limits of system codes with respect to scaling then are listed.

Then, the process of applying the code to the reactor is considered by building a reactor-transient input-deck. The development of the input deck, the criteria to apply, the qualification of the input deck at a steady-state level and a transient level are described, and also the role of the Kv-Scaled calculation.

The "scalability issues" are treated by considering issues related to closure laws, to code development, to validation and verification, and finally to applying the code to a reactor transient.

The scaling and uncertainty quantification of system codes is treated for code application in a BEPU approach.

Finally, possible Scaling Roadmaps for the application of codes to nuclear reactor safety issues are proposed.

4.1 Code merits and limits with respect to scaling

4.1.1 Scaling in the development of code models

SYS TH codes model the thermal-hydraulic physical system, and other related coupled systems. The thermal-hydraulic system can be either the cooling circuits of a nuclear reactor, or the circuits of a test facility, which are simulated by solving systems of equations. The thermal-hydraulics of the cooling circuits generally is treated by a generic method used for all components. However, some specific components having a particular geometry require specific thermal-hydraulic models. Thermal-hydraulics also is coupled to non-thermal-hydraulic systems that also are modeled in the SYS TH codes. Appendix 4.1 gives an overview of the generic thermal-hydraulic model, some specific models, and the non-thermal-hydraulic systems.

The best-estimate system codes were designed with two main objectives:

- Being able to model correctly all important phenomena with sufficient accuracy for safety analyses;
- Applying the necessary simplifications to make the code applicable for solving nuclear reactor safety and design problems.

As shown in Appendix 4.1, several successive simplifications are made in the process of development. Each simplification may result from process ranking or some scale considerations that may justify the simplification. There may be terms in the equations that are neglected after evaluating their order of magnitude, there may be processes that may not be modeled after evaluating their impacts compared with that of others, and there may be non-dimensional numbers which may not be considered in some closure laws after selecting the most important ones. Some of these scaling considerations are listed:

- Using time-averaged equations to filter out turbulence and two-phase intermittency (typically the time between two bubbles) is derived from scaling analyses which showed that the time scales of the dominant phenomena of interest are, in most cases, larger than the time scales of turbulence and two-phase intermittency.
- Using O-D or a lumped model in some component may result from the evaluation that the internal 3-D-velocity field does not play an important role, and that transfers between walls and interface may reasonably be well predicted without solving 3-D equations. This may be the case when the pressure field is quasi-hydrostatic, and when natural convection rather than forced convection heat transfers is used.
- Using 1-D cross-section averaged equations for many components results from the evaluation that the flow is quasi-unidirectional, and that the transverse profiles of flow parameters are quasi-established, but not changing too much with the abscissa, so that established radial- transfer coefficients (for wall-friction and wall-heat transfers) may be used in the radial direction.
- Using porous 3-D approach that may represent only large-scale 3-D effects in a core, results from

considerations that smaller-scale 3-D-effects are less important ones.

– A rather coarse nodalization may be used for modelling thermal-hydraulics, even in the core. This choice results from considering that the figures of merit in safety analysis do not exhibit strong local variations that would require a finer resolution. In this situation, the peak clad temperature may change by degrees with the size of a mesh, not by tens of degrees or more.
– Using a 1-D heat conduction in heating structures and in passive solid structures is a good approximation when the heat flux in the radial direction perpendicular to the fluid-solid surface is much larger than the heat flux in the other directions. In presence of a high axial heat flux at a quench front, a 2-D conduction calculation is available in most system codes for Reflooding.
– Scaling considerations are important in flow-regime maps since transitions criteria often depend on the geometrical scale.
– Scaling considerations are important in every closure law since they use non-dimensional numbers and some may include geometrical scales.

4.1.2 Scaling in code verification

Verification is a process to assess the code's correctness and the numerical accuracy of the solution to a given physical model defined by a set of equations. In other words, verification is undertaken to show whether the equations are solved correctly by the code. Thus, the relationship of the results of the calculation to the real world is not an issue in verifying the code. Simply speaking, verification deals with mathematics and data processing. In a broad sense, the verification is performed to demonstrate that the design of the code's numerical algorithms conform to the design requirements, that its logic is consistent with the design specification, and that the source code conforms to programming standards and language standards.

Appendix 4.2 describes the SYS TH code verification activities.

Since Scaling is related to the physics of a problem to solve and Verifications deals with the numerics of a code, there should not be strong relations between them. Anyway verification includes many aspects that may have some relation to the scaling issues.

– Space and time scales: During the PIRT analysis of a problem, some important flow processes are identified which play a dominant role. These processes have time- and space-scales that may be small or large, depending on the case. For example, a time resolution of 1 s or even 10 s may be sufficient to describe important phenomena in long- and slow-transients, but very short time scales (10-4 s, 10-5 s) are related to the propagation of a pressure wave in a circuit after a break opening if the associated calculations of mechanical loads are the processes of interest. Also, some minimal space-resolution may be required to investigate other processes. The requirements for time and space resolution apply both to the physical modelling and the numerical solution. The physical modelling should describe the physics with the necessary resolution, and the numerical scheme should be able to solve the physical model with a sufficient accuracy and stability. For example, the accuracy of transport processes has to be consistent with respect to some acceptance criteria. This may be important for the transport of a temperature front (e.g. a positive reactivity feedback due to the flowing of cold water into the reactor's core by overcooling in the steam generator during a break in a steam line) or for the transport of a boron concentration (e.g. a positive reactivity feedback due to the flowing of un-borated water into the reactor core in case of small break loss-of-coolant accident). Then, when the required minimum space and time scales are defined, verification should check that the numerical scheme can resolve these scales.

– *Accuracy and scale:* The numerical scheme solves the equations with an accuracy that depends on its mathematical properties. When solving a similar problem in a reduced-scale test or in a reactor, there may be a unique solution if the two problems are actually "similar". A frictional pressure loss in an adiabatic pipe may depend only on a Reynolds number. In a reduced diameter pipe, one may fulfill the Reynolds number similarity by increasing the velocity to compensate the pipe's lower diameter. Then, the equations to be solved are identical in a non-dimensional form, and the accuracy of the solution should be the same at both scales. However, since equations are not written in a non-dimensional form, the accuracy may differ at different scales. One of the contributors to uncertainty is mesh/node size used in the plant and facilities of reduced size where time is preserved.

– *Scalability of coding errors:* Coding errors may induce any type of error, and may include errors that affect the code's scalability. Imagine a coding error that would be quantitatively small in reduced-scale test simulation, and which would have a higher impact at reactor scale. Also, imagine numerical errors that may have a larger effect at the reactor scale than at a reduced scale. This seems possible, since no system-code writes non-dimensional equations.

Verification should take care of these three scaling issues in the following ways:

– checking that the numerics solve properly the required time- and space-scales,
– measuring accuracy versus scale,
– tracking scale-dependent coding errors.

4.1.3 Scaling in code validation

Validation is the process for assessing the adequacy of the physical models of the code. Physical models include some first-principles laws that do not require any validation, and many closure relations that are simplified descriptions of the flow processes, and which require validation. The main aspects of these physical laws and closure relations of current SYS TH codes first are summarized, focusing on the thermal-hydraulic models. However, non-thermal-hydraulic models such as neutron kinetics, fuel thermo-mechanics, and hydrogen production also may involve simplifications and closure relations to be validated. Then, some characteristics of the validation matrices are given with the selection criteria, the role of different kinds of tests, and the way code results are analysed. The content of validation report is defined and the role of validation in code uncertainty methods is presented. The function of sensitivity tests during the validation process is explained. The development and qualification of nodalization is addressed, and the relations between validation, user effect and user guidelines are discussed.

Code validation process and scaling in code validation process are intimately related. The words 'validation for scaling' (or validation with respect to scaling) can be introduced here. The following are the objectives of validation for scaling:

– ***Scalability of each closure law***: Each closure law was developed from the analysis of some experimental data, and after a scaling analysis that selected the relevant non-dimensional numbers which may play a role in the process modeled by the closure law. For example, a convective heat transfer in a single-phase fully established heated pipe-flow may be modeled by the relation between the Nusselt-, Reynolds-, and Prandtl-numbers. The validation on SETs should be able to demonstrate the scalability of each closure law, i.e. demonstrate that no other non-dimensional number (other than those present in the closure law) plays a significant role in the process modeled by the closure law. The validation on SETs should be able to demonstrate that the quality of predictions sensitive to a closure law does not depend on the scale of the SET, and on the value of all dimensional numbers present in the correlation. For example, the convective heat transfer in a single-phase fully established, heated pipe-flow should be validated in the whole range of Reynolds number encountered in the reactor situation of interest. If it is not validated in the whole range, at least it should be shown that it takes correctly the Reynolds effect in a rather large range of Reynolds number.

– ***Scalability of each module of the code used for the situation of interest***: O-D, 1-D, 2-D, or 3-D modules are used to model the reactor's components to perform transient analyses. They include many simplifying assumptions that may affect to some (a priori unknown) degree the quality of predictions, depending on the situation of interest. A simplification of a 3-D flow (all flows are 3-D) in a 0-D or 1-D model may result from a scaling analysis that established that the main phenomena may be predicted with a 0-D or a 1-D model. The validation on both SETs and IETs may demonstrate the correctness of the choice of a specific module for a specific component, and/or situation.

– ***Scalability of the code for a reactor transient***: Most reactor transients of interest are simulated in IETs at reduced scale. IETs were scaled following some scaling criteria. To demonstrate the scalability of a system code with respect to a transient, it should be demonstrated, at least, through validation on IETs that the code properly predicts the main parameters of the transient in different IETs having different scale factors.
– ***Scalability of the code for some scale distortions****:* The typical scaling basis for the design of IETs is fixing a hierarchy in scaling factors. Priority is given in the design to those factors assumed to minimize the scaling distortions between the IET transient and the reactor's transient. However, there are some well identified distortions. The impact of a distortion in an IET is studied by using an appropriate SET. Then, the resulting validation of the code may demonstrate that it predicts the distorted- and non-distorted-phenomenon with the same quality. For example, the design of an IET may not be representative of the reactor for a flooding limit in a particular component because the scaling does not encompass the non-dimensional numbers that control the flooding limit. The flooding limit may be investigated in a SET with the reactor's geometry and with a distorted geometry, and the code may be validated on both of them.

4.1.4 Limits of system codes for scaling

4.1.4.1 Limits related to space- and time-averaging

System codes do not predict small scale thermal-hydraulic phenomena due to space averaging, and cannot predict all the small time-scales associated with turbulence and two-phase intermittency. This inherently limits the scaling capabilities of the system codes. If phenomena having small time- or space-scales play a major role in a transient, they must be modeled in a way that accounts for all the parameters that control or influence the process. There may be small-scale local geometrical details that influence these small-scale flow processes. In such cases, the code user should enter the necessary information so that the code can predict the process correctly. Examples are the following ones:

– Singular geometries induce singular pressure losses that depend on the local geometry. The system codes do not predict these losses and the user must enter in the input deck the form loss coefficient, depending on the geometry.
– Singular geometries may affect flooding limits and CCFL that depend upon the local geometry. The system' codes cannot predict these flooding limits, and the user must use a CCFL option and enter in the input deck the parameters of the local flooding limits taken from representative experiments.
– The geometry of a break influences the value of the critical flowrate. The available 0-D or 1-D models in the system's code for predicting the flowrate from a break cannot predict all the effects of the break's geometry.
– Many reactor components have small-scale characteristics that influence their macroscopic behaviour, and the code cannot predict it without having (as input) the information coming from the measured characteristics of the components. This typically is the case for separators, dryers, pumps, turbines, valves, safety valves, control valves, check valves, flow limiters, and spray cooling. Any new designs of such components require experimental characterizations of their macroscopic behaviour.

4.1.4.2 Limits related to the dimensions of the model

Using an O-D (or lumped) model, 1-D models, or a porous 3-D approach involves simplifying a complex 3-D flow; using a 1-D heat conduction in heating structures, and in passive solid-structures is an approximation for a more complex 3-D conduction. There may be many situations for which the degree of approximation is reasonable, and does not affect the code's capability to solve safety issues. However, there may be situations where the modules of a system code do not allow predicting a specific phenomenon (in other terms there are situations which cannot be predicted correctly by any of the code modules).

Some inherent limits of the 0-D-, 1D-, and porous 3-D-modules are given below.

4.1.4.3 Specific limits related to the 0D models

O-D- or lumped-models are used in some reactor components, such as the Vessel Upper Head, the Lower Plenum, the Upper Plenum, the Pressurizer, the inlet- and outlet-SG headers (also called mixing chambers or inlet and outlet plenum), and the SG steam dome. The choice of the O-D or lumped model may result from the evaluation that the internal 3-D velocity field does not play an important role, and that wall and interfacial (energy, mass and momentum) transfer processes can be reasonably well predicted without solving 3-D equations. This may be the case when the pressure field is quasi-hydrostatic, and when natural circulation heat-transfers take place. But the choice also may result from the absence of a better module capable of resolving the 3-D flow in an open medium. In some specific situations, they may be important phenomena that cannot be predicted by any of the available modules. Examples of limitations which are not necessarily, in principle, scaling related, but are affected by the size of components or systems, are the following ones:

- Temperature stratification in some components may be destroyed by some local flow configuration, entailing a sudden increase of the interfacial heat-transfers (sudden high condensation-rate).
- Temperature stratification in some components may result from natural convection cooling along the cold walls or heating by hot walls, and the resulting temperature field depends on turbulence mixing. No 0-D, 1-D, nor porous-3-D model can predict the situation. In containment compartments, this may also include the stratification of mass concentrations of air, steam, and hydrogen.
- There may be situations in SG with very low pressure-differences between the inlet and outlet headers wherein the flow may be positive in some tubes and reverse other tubes. Such situations also may depend on the mixing phenomena within the inlet header that cannot be predicted by any of the 0-D-, 1-D-, or porous-3-D models.
- O-D modules sometimes are used for multi-connection components with rather high velocities, such as the upper part of a downcomer connecting all the cold legs with the upper head and downcomer. Predicting the pressure field and pressure losses for all possible flow directions at each connection is far beyond the capabilities of such lumped models.

4.1.4.4 Specific limits of 1-D models

A 1-D model is the basic model of system codes since, in the most important components of a LWR (core, cooling loops, heat exchangers, or steam generators) there is a privileged direction of the velocity field so that averaging over the cross section of the duct is natural, while keeping only a momentum equation projected along the flow direction. It does not mean that the flow is 1-D since all flow parameters (velocity, void fraction and fluid temperatures) may have large transverse variations. But it means that transport is along the direction of flow direction while the diffusion processes mainly are radial. The radial transfers appear in wall transfers and the interfacial transfers of mass, momentum and energy.

One strong assumption is that all these transfers can be expressed as function of the principal local variables. These functions are often algebraic functions and may be also function of local derivatives of principal variables (added mass force, pressure at the interface, grad-α term). Limitations are associated to this strong assumption:

- Closure laws cannot be functions of the abscissa along the flow in the duct (because of the numerical solution method), and cannot take into account of the establishment of the flow: Wall friction and heat-transfer coefficients consider established flow conditions.
- Historic effects on the turbulence level and on the two-phase flow regime cannot be taken into account. This is very limitative in expressing an average bubble size in bubbly flow, or an average drop size in annular-mist-flow. Coalescence, break-up, nucleation, collapse, vaporization, and condensation may affect the size of the bubble or drop. All these phenomena can be taken into account in a transport equation for bubble (drop) number density, or for interfacial areal density, but modelling the size of the bubble or drop using algebraic equations of the local main variables is only possible in very particular cases, e.g. when all bubbles or drops reached their limiting break-up diameter given by a Weber number.

There are situations where the velocity field is not exactly unidirectional, in particular when the effects of natural convection play a role.

Other limitations exist due to the lack of modelling of axial diffusion and axial dispersion of momentum and energy in most 1-D models (some options to model axial diffusion-dispersion may exist in some system codes) for some applications where it may become a sensitive phenomenon, e.g. boron dilution or the mixing of hot and cold water in an MSLB.

4.1.4.5 Specific limits related to porous 3-D models

Porous 3-D models are used in 3-D Pressure Vessel Modules available in several system codes. The aim is to represent only large-scale 3-D effects such as

- Radial-power profile effects in a core.
- Azimuthal heterogeneity of flow conditions in an annular downcomer, due to repartitioning of the cold leg flows. For instance, this plays an important role in the refill phase of a LBLOCA.

Constitutive relations used in 2-D and 3-D models generally are extrapolated or simply taken from 1-D models:

- Wall-heat transfer coefficients do not depend on the direction of the coolant velocity with respect to the fuel rods.
- Wall frictions and pressure losses may be different in axial- and radial-flow but there is no effect of the radial velocity upon the axial pressure loss and no effect of the axial velocity upon the radial pressure loss.
- Flow-regime maps are the same as in 1-D models, although radial velocity may probably affect the bubbles' size, droplet entrainment, and deposition.
- Interfacial friction is isotropic in a non-isotropic medium.

In most system codes adopting 3-D models there is a lack of modelling of turbulent diffusion and dispersion. Component codes that are used for CHF investigations have models for radial diffusion and dispersion, which have a significant effect on the occurrence of CHF. However, the models are adapted to sub-channel analysis, and they should be extended to larger space-filtering as used in applications of system code.

4.1.4.6 Specific limits related to 2-D models

Porous 3-D Pressure Vessel Models of system codes usually apply 2-D descriptions of the RPV annular downcomer with only one mesh in the radial direction to predict the 2-D effects during the refill phase of a LOCA. This configuration is consistent with the limited information available on two-phase flow in the downcomer. UPTF tests are used to validate these 2-D downcomer models because they provide global information (flowrates), but also some temperature maps which give an indication of the phase repartitioning. However, in the vertical- and azimuthal-directions, there may be turbulent transfers that one cannot model in the whole range of flow regimes.

4.1.4.7 Limits related to flow-regime maps

Every flow regime has its own internal structure and its transfer mechanisms. Flow-regime maps provide the necessary information on interfacial structure and interfacial area based on experimental observations that, together with some theoretical basis, allow a mechanistic modelling of interfacial transfers.

So, it seems natural to use a flow pattern map in a code and to develop correlations for mass momentum and energy transfers that depend on the flow pattern. This usually is done via flow-regime maps wherein specific two-phase flow patterns are identified as functions of input data, such as superficial gas- and liquid-velocities, flow rates, or more complex dimensionless parameters. The highly empirical flow-regime maps are based on a large amount of measured data for different fluids, different conditions of vertical- and horizontal-flow and different pipe diameters. Typical examples are the flow pattern maps proposes by Baker (Baker, 1954), Mandhane et al., 1974, and Taitel & Dukler, 1976. Nevertheless, all these flow regime maps are valid only for steady state, quasi-steady state, fully developed, or quasi-developed flow conditions, although rapid transients and non-established flows also occur in nuclear reactors under accident conditions. With these restrictions in mind, it might be justifiable to further simplify the flow maps as usually it is done in all the codes.

As an example, the flow regime map for horizontal flow, as used in RELAP5/MOD3 (RELAP5 Code Development Team, 2001), is shown in Fig. 4-1. Major selection parameters are the void fraction, α_g and the total mass-flow density, G_m.

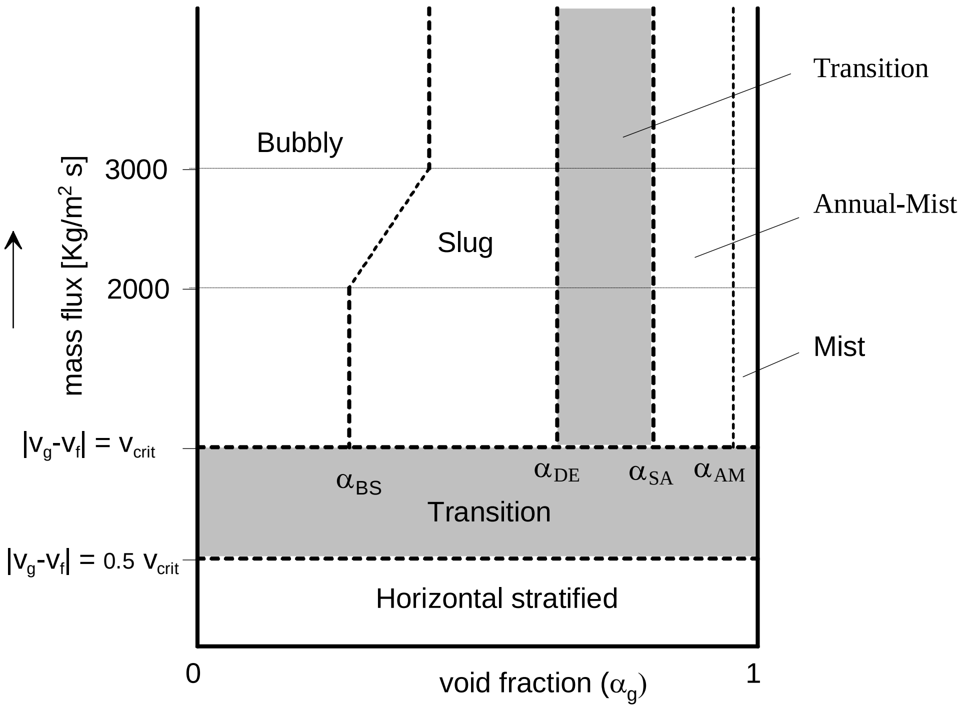


Fig. 4-1 - RELAP5 flow regime map for horizontal flow.

Specifically modelled are the regimes, such as bubbly, slug annular-mist, mist, and stratified. The transition criteria between the regimes also are algebraic relations between flow parameters. To avoid discontinuities, some transition regions are included wherein all the parameters are interpolated from the boundaries of the adjacent flow regimes. This only provides artificially smooth transitions, but cannot take into account the relaxation-time constant associated to the flow process responsible for the transition.

Transition criteria

Horizontal stratified flow conditions are assumed to exist for

$$G_m < 3000\, kg/m^2 s$$

And

$$\left| v_g - v_l \right| < v_{crit}, \qquad (4\text{-}1)$$

with the critical velocity according to Taitel & Dukler, (1976)

$$v_{crit} = \frac{1}{2}\left[\frac{(\rho_l - \rho_g) g \alpha_g A}{\rho_g D \sin\theta}\right]^{\frac{1}{2}} (1 - \cos\theta) \qquad (4\text{-}2)$$

The limiting mass flux is a criterion that is not formulated with non-dimensional numbers, and arguably cannot predict any scaling effect.

Similar flow regime maps are applied for vertical flow, including pre-CHF and post-dry-out conditions, high mixing-flow conditions in pumps, and for mixing sub-cooled ECC water with near-saturated steam, and the resulting condensation processes.

These are simple extrapolations from adiabatic- to non-adiabatic-conditions, although heat- and mass-transfers may significantly affect the flow regime:

– Flashing flow creates many small bubbles and accelerates flow that increases turbulence and enhances the process of break-up of bubbles;
– Direct contact condensation may induce large-scale instabilities and condensation-induced water-hammer.

None of the available maps consider these two important effects of interfacial transfers on flow regime.

The most important transitions correspond to changes in the interfacial area of several orders-of-magnitude, e.g. at the onset of droplet entrainment, or when phase stratification occurs in a bubbly flow in a horizontal pipe.

Several models are used for the onset of droplet entrainment:

– Steen-Wallis model:

$$J_{g0} = 2.1*10^{-4} \frac{\sigma}{\mu_g}\sqrt{\rho_g / \rho_l}$$
$$E = 1 - \exp(J_{g0} - J_g) \quad (4\text{-}3)$$

E being the entrainment rate,

– Ishii-Mishima model:

$$E = \tanh(7.25e^{-7} We_e^{1.25} \mathrm{Re}_l^{0.25}$$
$$We_e = \frac{\rho_g J_g^2 Dh}{\sigma}\left(\frac{\Delta\rho}{\rho_g}\right)^{1/3} \quad (4\text{-}4)$$

– Or a limiting value of the Kutateladze number.

The main reason for codes not having converged to the same correlation probably is that their validation could not identify clearly which model was the most appropriate one. However, these models do not have the same dependence upon the geometrical scale, D_h, and no data for large diameters are available.

The following are the main limitations of using such flow regime maps:

– Range of validity: There are no measured data on the flows of high-pressure steam water flows to validate the flow maps in such reactor conditions. Also, data from large diameter pipes (e.g. 1 m) are very limited.
– Flow geometry: Only very limited measured data of flows in various complex geometries (rod bundle, lower plenum, upper plenum, annular downcomer and SG headers) are available, although geometrical effects are likely to be significant. Also, the effects of some singularities are not taken into account.
– Steady and established flows are necessary to establish such maps, and they are extrapolated in transient conditions or non-established ones. Historic effects and relaxation time-constants associated with regime transitions are not accounted for.
– Effects of interfacial heat and mass transfers on the flow regime map are not taken into account, although there may be huge effects from them.

4.1.4.8 Limits related to scaling each closure law

Closure laws in system codes may be either purely empirical or mechanistic, or semi-empirical. The scalability of the closure law depends on their nature:

- A purely empirical correlation is a best fit of experimental data wherein the quantity to model is expressed as any function of the principal variables. It can be very accurate within the domain of experimental investigation, but extrapolating beyond it is very dangerous. The scale-effect may be taken into account by a diameter or shape effect (rod bundle versus pipe geometry in the CHF look-up tables). Scale extrapolation beyond the domain of scales where data were used is forbidden (or, in any case questionable).
- The phenomenological- or mechanistic-approach consists in assuming a governing physical mechanism. The correlation then is derived theoretically without anything coming from experiments. This approach properly accounts for the scaling effects, as far as physical assumptions are valid. Purely mechanistic models are only very few exceptions in current system codes, and most models include some degree of empiricism.
- Semi-empirical models rely on some governing physical assumption, but retain some free parameters to adjust the experimental data. This semi-empirical approach is the most frequently used one in the current system codes. Even with this last precaution, the correctness of extrapolation of data beyond the qualification domain is not guaranteed. New effects, which are not present in the model, may become important in another range of parameters. Experience showed that two-phase thermal-hydraulics contains myriads of phenomena, making it difficult to generalize any theoretical model. Interpolation often is possible but scale extrapolation is troublesome, even with physically based models.

Other limits for scaling:

- Closure laws are based on established assumptions on flow that are no longer valid in many situations encountered in reactor circuits. Some pipes in cooling loops may have a short length-to-diameter ratio. Transitions in flow regimes in two-phase flow have time- and length-scales that are not taken into account.
- Mechanical closure laws for wall- and interfacial-friction do not depend on the interfacial heat- and mass-transfers although these transfers may have significant effects.
- Two-phase flows are complex ones wherein many non-dimensional numbers may play a role in every transfer. Closure laws use a few non-dimensional numbers, and some of them may include geometrical scales. It is difficult to guarantee that other non-dimensional numbers – that may include geometrical scales – do not play a significant role in the whole domain of simulation.

4.1.4.9 Limits related to non-modeled phenomena

All codes model some of the phenomena and either ignore or neglect others. System codes neglect many phenomena, as detailed in the subsections above.

Most of those neglected already were mentioned since they are related either to time- and space-filtering, to closure laws, or to the intrinsic limitations of 0-D, 1-D, or 3-D modules.

In the history of the current best-estimate system codes, there are a few examples of phenomena that first were ignored and later implemented when their impact was found to be significant.

For example, phase separation at a break in a horizontal pipe plays an important role on the quality of the flow going to a break, depending on the relative position of the free surface of a stratified flow with respect to the position of the break (either vertical upward, vertical downward, horizontal,). Then SETs were devoted to validating of these phenomena and to developing models that were implemented in the system codes.

Other specific phenomena also were also added, such as CCFL in complex geometries, and direct contact condensation in the vicinity of an ECCS injection.

Seemingly, many other non-modeled phenomena may influence reactor transients. If these phenomena exist in both IETs and reactors, system's codes may give have an estimation of the impact of the phenomenon on the transient; this may be taken into account in the global uncertainty of the code's predictions. If the non-modeled phenomenon plays a larger role at the reactor scale than in scaled IETs, there is a problem in scaling.

4.1.5 Code-up-scaling capabilities

Codes that are validated on some scaled SETs and IETs may be able to predict the phenomena of interest at the reactor scale, provided that some conditions are satisfied. This is called their up-scaling- or scaling-up capabilities. Among the main conditions to satisfy are the following:

a. The code has been validated on the transients of interest performed in scaled IETs that represent the main phenomena of the transient, as identified in a PIRT, and predict well qualitatively and quantitatively the main phenomena.
b. The code was validated on the transients of interest performed in several scaled IETs at different scales, and the code predicts their effect or the absence of effects.
c. The code has proved that closure laws have a good up-scaling capability by validating all important phenomena at local- or component-scale against several SETs at different scales.
d. The code has proved that closure laws' validation domain cover the entire prototypical thermal hydraulic range of interest.
e. Since the scaled IETs necessarily have some scale distortions, the code should be able to predict correctly the distorted phenomena. This may require a validation of the distorted (in IETs) phenomena in non-distorted SETs.
f. The code is used in reactor simulations with the same numerical schemes and numerical options as were used for validating SETs and IETs.
g. The code is used in reactor simulation with the same set of equations and closure relations - and the same empirical constants - as were used for validating SETs and IETs.

The code is used in reactor simulation with a nodalization, and a time-step as close as possible to those used for validating SETs and IETs relative to the physical situation of interest, and following all recommendations on the best nodalization and time step that were derived from validation studies, and that may be given in the User's Guidelines.

4.1.6 Preliminary calculation to verify scaling laws

The design of any new facility is based on scaling analyses that lead to geometric parameters that define the test facility. However, before constructing an experiment facility, evaluating the scaling using computer codes – in particular the system's thermal-hydraulics codes – could increase the confidence level of scaling through identifying errors and missed phenomena in the components. For a complex system, such as an integral test facility, many components and local phenomena need to be scaled. There always is the possibility of missing phenomena in scaling a large system, and inevitably, there are scale distortions.

Computer models can be constructed for a prototype and a scaled model to simulate their responses, and to evaluate similarities between the two. In some cases, a direct comparison is reasonable. However, it is important to establish that the code is applicable for the plant and the scaled facility for postulated transients: this should be established at least by validating the closure relations of the code on a suitable SET database, including suitable range of variations for key variables.

It is expected that the scaled facility always will have design- or construction-compromises compared to the prototype. For example, in an integral reactor-test facility, electrically heated rods are used to simulate the core. Though the power output can be accurately scaled, the heat flux and the temperature of the core fuel rod surface may be compromised if material properties are not properly represented, or if heat transfer coefficients are not similar. Heat loss to the environment is another example of additional scaling distortion. To study these distortions, a triad method, Ransom et al., 1998, was devised for relating the scaled experiment to the conditions of the prototype system. This method, somewhat connected with the Kv-scaled method discussed in section 4.2.4, provides a means for substantiating the overall scaling philosophy. The method is based on using three separate, but related computer models (Fig. 4-2). The three models are: (1) the prototype; (2) an ideal scaled experiments; and, (3) the scaled experiment. These three models are used to investigate the degree to which qualitative- and quantitative-similarities are maintained between the three systems for a particular process, as depicted in the figure below.

The prototype model is a full-scale, best-estimate model of the system, and is used for estimating the baseline response of the system for a postulated accident, or a particular physical process. The ideal scaled model then is derived from the prototype in such a way that a homologous relationship to the prototype is maintained throughout the transient, or particular physical processes for all significant safety-related parameters, i.e. the distribution of the fuel' temperature distribution, the flux of surface heat, and the rate of energy release, as well as the similar loop pressures and flows. This task requires appropriately scaling the core's geometry and the fuel properties, in addition to the geometric- and kinematic-scales used in the experiment. The third model corresponds to the test facility which incorporates any non-typicality necessitated by practical limitations in its design, such as the need to meet ASME pressure-vessel codes, materials limitations, heater design, instrumentation needs, and more.

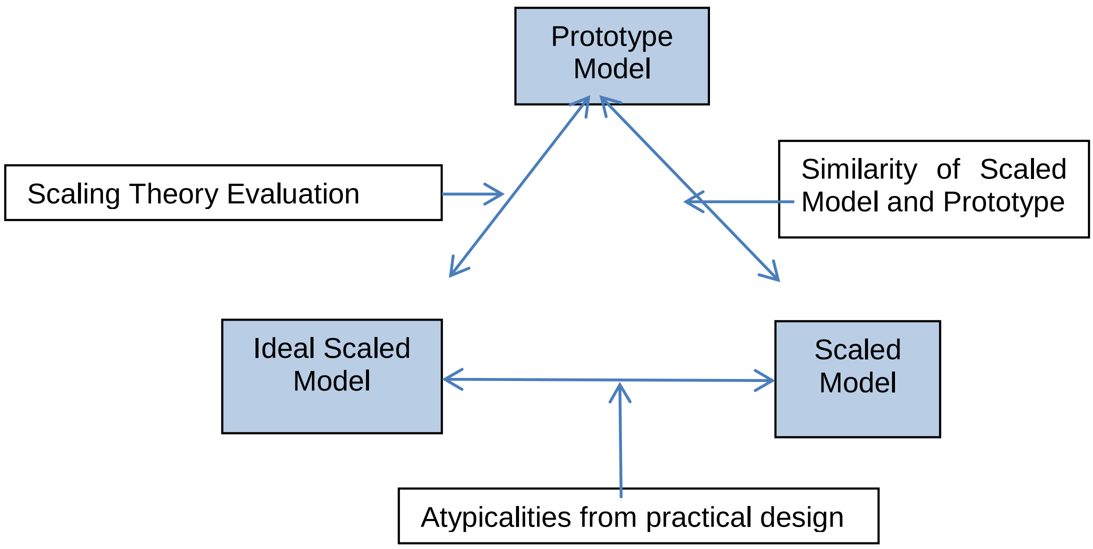


Fig. 4-2 – The triad method.

The benefits of this triad of models are to ensure the issues of homology are examined: (1) The response of the prototype and the ideal scaled model are compared to assure the preservation of qualitative-, and, to the degree feasible, quantitative-similarity, and, (2) the effect is evaluated of any experimental non-typicality, such as physical configuration, heat loss, and the real valve-opening times. This Triad scaling-evaluation method was applied to the scaling and design processed of Purdue University's Multi-dimensional Integral Test Assembly, PUMA, Ransom et al., 1998. The method highlighted some missed local phenomena and components, and proved to be a necessary step in the scaling and the facility's design.

The scaling of PUMA is well documented, Ishii et al., 1996. The volume ratio was chosen as 1/400 and height ratio was at 1/4. This resulted in the flow area's scaling ratio being 1/100. The basic principles of kinematic, Froude number, and energy-density similarity then were used to establish the scale ratios for time (1: 2), velocity (1: 2), and power (1: 200).

The RELAP5 simulations of main steam line breaks were used to compare the prototype and the ideal scaled model. Initial calculations showed that the scaling ratio of the reactor's response time was not at, or close to, 1: 2 as designed. In other words, the pressure decrease in the scaled model is much faster than what was expected. Due to this observation, further investigations were undertaken; they revealed that when the flow was choked, the areal scaling ratio needed special consideration. At the choking plane, the flow velocity is prototypic because the Mach number for a choked flow is unity, and the sound speed is a function of the prototypic pressure, temperature, and fluid properties. This condition resulted in a flow area scale ratio of 1: 200 at the location of choked flow, rather than 1: 100 as determined from considering volume and power.

After correcting the scaling of the choked flow, values of the time-scale of the ideal scaled model were about half those of the SBWR prototype, which is consistent with the scaling theory. Figure 4-3 compares the pressure transient of the prototype, the ideal scaled model, and PUMA. In the PUMA tests, the experiment started as the system's pressure dropped to 150 psia after the initial blowdown. Figure 4-4 shows the fuels' temperatures. These results reveal that the values of pressure and temperature values are the same at corresponding points in space and scaled time, and a homologous thermal relationship is obtained. Both qualitative- and quantitative-similarities are evident.

4.2 The process of building a reactor transient's input deck

The first principles-derived balance equations of mass, energy, and momentum, the models, and the correlations are relevant to scaling analysis as well as within the processes of developing and validating a SYS TH code. All of this is discussed in previous sections and chapters. When entering into the process of applying the SYS TH computer code to a safety analysis of nuclear power plants, and when attempting to validate the code, an 'additional' software product is needed, viz., the nodalization, or (transient) input deck, or idealization (according to Canadian specialists).

Hereafter, the additional software is characterized with the word 'nodalization' the use of which has spread during the last four or five decades.

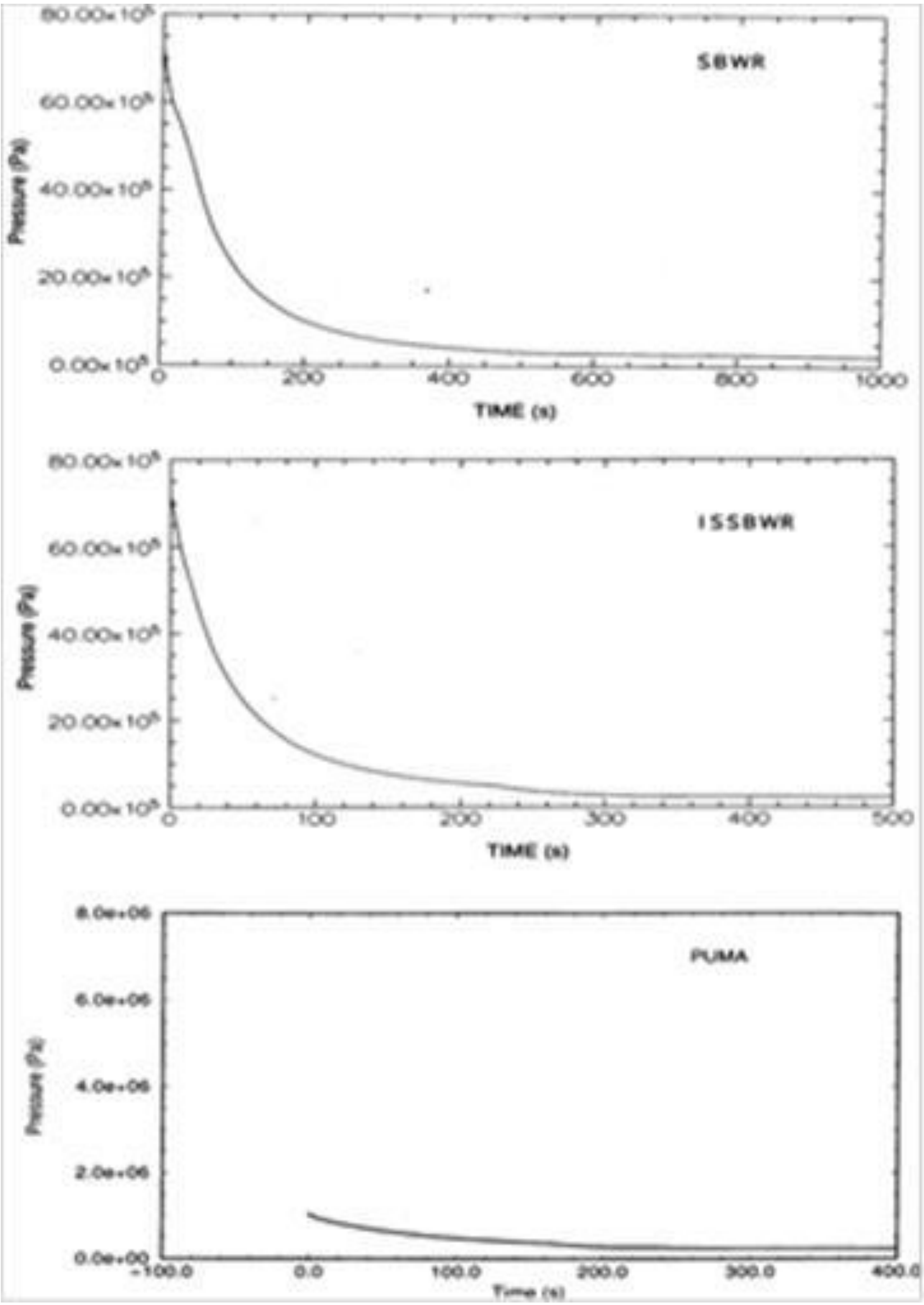


Fig. 4-3 – Pressure comparisons for the prototype SBWR, an ideal scaled SBWR, and PUMA.

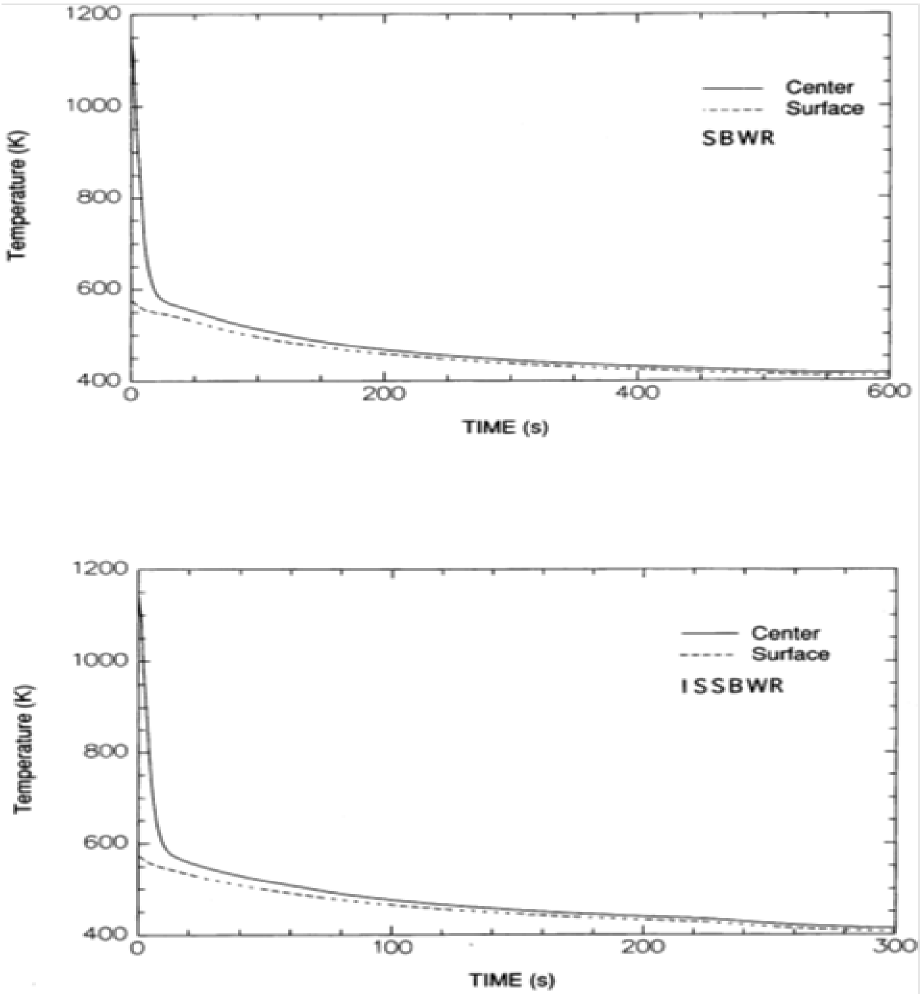


Fig. 4-4 – Fuel temperature comparisons for the prototype SBWR and an ideal scaled SBWR.

Nodalization constitutes the connection between the code and the physical reality. Nodalization typically is produced (or developed) by the code-user, ‘the analyst’ or, ‘the group of analysts’. The nodalization can be seen as the result of a brainstorming by the group of analysts involving their knowledge of the code features, of the physical system to be simulated, as well as of the expected transient-scenario objective of the calculation. The computational resources available, i.e. computer power and time needed for one calculation (this is called Central Processing Unit time or CPU time) also play a role when setting up nodalizations. The purpose of the analysis (e.g. safety calculation, design optimization, supporting a sensitivity study, and code validation) and the human resources available also are also considered.

Nodalization is an indispensable component for code applications. As a feature of nuclear system’s thermal-hydraulics (as a possible difference from other disciplines) the results from code applications that are relevant to NPP technology cannot be obtained only via the code. For instance, nodalization is needed to determine natural circulation mass flow-rate in a PWR system: this quantity cannot be calculated without nodalization and its value is largely affected by the details of nodalization, including the choices typically needed from a group of analysts. In different terms, the code models alone are insufficient to determine the natural circulation’s flow-rate, but a reference system is needed along with a nodalization.

The role of the nodalization is clarified in the following statement: If an excellent code is developed and properly qualified for an assigned application, and a poor nodalization is used, low-quality results are expected. This situation may reveal possible flaws in the framework of the code’s validation: poor comparison with experimental data is associated with inadequate modelling. So, following the application of a poor nodalization, an un-needed attempt is made to change the code models. The new code capability may become consistent with the poor nodalization, so that the process of validating the code is severely delayed.

The connection with scaling

The connection between nodalization and scaling is embedded into the definition of scaling: There is a need to prove the validity of SYS TH codes against scaling, and then, there is the need to develop input decks for differently scaled experimental facilities. More details are given in the following key sample topics, which are at the origin of the relationship between scaling and nodalization:

- The length-over-diameter ratio (L/D) is associated with meshes or “control volumes”. On the one hand, the L/D ratio constitutes a key parameter for developing nodalization and for its numerical solution; on the other hand, it is impossible or impractical to preserve the L/D when setting up nodalizations for differently scaled facilities. Unavoidably, the results of calculation are affected by the L/D: it must be proved that the impact of different L/Ds is tolerable, i.e. within acceptable error.
- Averaging region. The averaging region, including the cross-sectional flow area or volume of the single node (or control volume) cannot be the same for NPPs and a typical facility scaled down by a factor 1/1000. It must be proven that the impact of the sizes of the averaging regions upon the numerical solution is tolerable. As an example, the size of the node upstream of a chocked section is used during a transient calculation, to determine the void fraction that typically constitutes an input for the Two-Phase Critical Flow (TPCF) model: changing the node’s size unavoidably causes a change in void fraction which then impacts the TPCF value. An analysis of nodalization scaling is needed to search for the optimum node (i.e. the size of which minimizes the error with respect to data) upstream a chocked section; some codes may use the alternative procedure of reducing the node’s size until a converged (and) acceptable result of the TPCF calculation is reached.
- Steady State, and flow region not fully developed. The concept of fully developed flow region, also connected with the L/D ratio (discussed above) is well established in fluid-dynamics. In the area of SYS TH codes, this is reflected in a paradox already identified by D’Auria et al., 1998: Qualified models and correlations embedded in a SYS TH code are developed based on experimental data gathered under the condition of steady-state fully developed flow; these conditions do not occur in applying the codes to the accident analysis of a NPP. The node density or the size of a control volume (hereafter termed as the numerical region) unavoidably creates a difference between the length of non-developed flow in reality and in the steady state and the fully developed flow-length inside the numerical region. This difference is affected by scaling (i.e. by the size of the system under consideration): it must be proved that the impact upon the numerical solution from nodalizations having different sizes of control nodes causes tolerable errors.
- The coefficient for local pressure drop at a geometric discontinuity including branching, i.e. the so called K-factor. It is clear that any NPP, as well as any facility simulating the overall NPP or parts of it, includes a variety of geometric discontinuities. At least two issues are generated, the first not necessarily associated with scaling: a) K-factors are not part of code development, and must be introduced into the nodalization by a code-user (who may use scaled data); and, b) no universal scaling criteria are suitable to determine the K-factors at geometric discontinuities. This may have large impact upon any transient scenario: the K-factor’ values, including those at flow-reversal conditions, largely affect the location of the stagnation point during a LBLOCA or the core-bypass flow during an SBLOCA. Therefore, a specific scaling-qualification for the K-factors part of a nodalization is needed.

What is discussed here?

The nodalization features as well as the processes for developing and qualifying a nodalization have been, and still are the subject of numerous papers in the open literature, e.g. Bonuccelli et al., 1993, and reports, e.g. D'Auria & Galassi, 1992, and even courses, e.g. Petruzzi et al., 2005, all of which contain a number of connected additional references.

It is not the intent here to repeat, or even to summarize the established findings. Rather, in the next three paragraphs, key nodalization topics are defined, and the connection with scaling is addressed.

4.2.1 Development processes and criteria

When developing a nodalization, the user has available the SYS TH code and the code manual, possibly User's Guidelines, i.e. the facility s/he needs to simulate (typically, not all the code manual's requested information is obtainable), and the expertise from his/her own, and from the group of analysts which the researcher is part of. A list of key-sample topics or issues that should be addressed, and are at the basis of developing the nodalization for a typical PWR-NPP (reference calculation) is the following:

- Rough total number of nodes for modelling the overall system & the reference length of a single node.
- Number of nodes in the core.
- Overall rough number of mesh points for the conduction heat transfer, including the number of meshes for simulating nuclear fuel.
- Use of parallel nodes (e.g. core, steam generator, downcomer), and/or of the three-dimensional capability of the code (if any).
- Simulation of core-bypass flow paths (that are of high importance in case of an SBLOCA).
- Simulation of branching-connections (e.g. cold leg to vessel, pressurizer surge-line to hot leg).

The consideration of these topics implies knowledge of the objective/target for the calculation, and of the phenomena to be simulated. The resources available (personnel, time-scale, and computer power) also may play an important role as was mentioned.

Scaling is relevant to address any of the listed topics, and similar ones. The fundamental motivation is the need for the analysts to test the code's performance in relation to the phenomena expected for the reference calculation, using the experimental data necessarily obtained at a reduced scale. Following the analysis of experimental data, the analysis optimizes the nodalization choices for the reduced-scale test facility, and finds unavoidable errors or differences between the measured and calculated variables. Other than accepting the 'unavoidable' errors (otherwise he has to refer to code developers for improving the codes), the analyst must address unavoidable scaling-related questions:

- What procedure and/or what key criteria have to be followed to pass from the scaled facility's nodalization to that of the PWR NPP?
- Recall in advance that accuracy is the 'known' error that characterizes the results of simulations of ITF experiments which were found acceptable, and uncertainty is the 'unknown' error which characterizes the NPP's (reference) calculation. Then the question is "Under what conditions the uncertainty can be derived from accuracy? or, Under what conditions the process of extrapolation of accuracy is feasible?"

The answer to the former scaling-question already requires a variety of considerations and recommendations, e.g. D'Auria & Ingegneri, 1997, and Ingegneri & Chojnacki, 1997. A key recommendation is to keep unchanged, as far as possible, the length of the nodes. In this case, referring to the comparison between the test facility and PWR-NPP nodalization, the L/D ratio also is distorted: the L/D ratio appears in any scaling study, and plays an important role in its numerical solution (see the discussion above).

The latter scaling-question can be answered from a suitable uncertainty study, as discussed in sections 4.4 and 4.5.

4.2.2 Qualification at steady-state level

Within the process of applying a code to analysing a transient scenario (either in the PWR NPP, or in any scaled facility) the needed step following the development of the nodalization is a demonstration of achieving 'stable' and stationary initial conditions. Typically, the assigned steady-state conditions values for pressure, pressure distribution, mass flow-rates, liquid levels, temperatures, mass inventory, and thermal power exchanges are compared with calculated values.

In most of the cases of interest, the values of the assigned conditions are the experimental data which are unavoidably characterized by errors. On the opposite side, the calculated data are affected, among the other things by not fully specified values, like the coefficients of the pressure drop at geometric discontinuities, the distribution of heat losses to environment, and small leakages from the pressure boundary. Therefore, differences are expected in the comparison between measured- and calculated-values under steady-state conditions. The question comes in relation to the amount of the difference that is tolerable.

Acceptability criteria for those differences were proposed by Bonuccelli et al., 1993, and applied to complex situations several times, e.g. Petruzzi et al., 2006.

The process of qualifying nodalization at steady-state level is not directly connected with scaling.

4.2.3 Qualification at the on-transient level

A nodalization, developed according to the process mentioned in section 4.2.1, and qualified according to criteria in section 4.2.2, still might be unsuited for applications. Comprehensive lists of input parameters for nodalization which need 'on-transient' qualifications should be created, Petruzzi et al., 2006. Typical examples are those parameters affecting flow- reversal through centrifugal pumps, and throughout the core-bypass paths and the critical flow at the break (e.g. the size of breaks in upstream nodes).

To assure the best values for these nodalization input parameters (Bonuccelli et al., 1993), two cases are distinguished:

- Case a): An experimental facility (either an ITF or a SETF) constitutes the target for the calculation concerned. In this case, experiments other than the target for the calculation exist. The user should demonstrate acceptability (see below) of the parameters in the list, following their comparison with the experimental data.
- Case b): The PWR-NPP constitutes the target for the calculation. In this case, a suitable data set to undertake on-transient qualification may not exist (with very few exceptions). Therefore, the analyst must use transient data measured during start-up or operational transients: thus, only a limited set of the list of parameters for on-transient qualification may be available for the assessment. Then the additional effort requested to achieve the best possible qualification is the so-called Kv-scaled calculation, as discussed in section 4.2.4.

To determine acceptability of on-transient calculation, i.e. referring to the comparison between time trends of a suited number of variables that characterize the transient, qualitative-, and quantitative-accuracy evaluation procedures, including acceptability thresholds must be used, Bonuccelli et al., 1993. Qualitative accuracy evaluation is based upon considering the phenomena: namely a check is made in relation to the presence of the same phenomena, including phenomenological windows in the calculated data set and in the experimental data set, (NEA/CSNI, 1989, NEA/CSNI, 1993, and NEA/CSNI, 1996). A quantitative accuracy evaluation can be pursued with the widely applied Fast Fourier Transform Based Method (FFTBM), Ambrosini et al., 1990, Mavko et al., 1997, and Kunz et al., 2002.

The process of qualifying nodalization at the on-transient level is not directly connected with scaling, with the noticeable exception of the Kv-scaled analysis, discussed in next section.

4.2.4 Kv-Scaled Calculation

In the framework of qualifying the NPP nodalization and meeting its quality assurance procedures, Kv-scaled calculations, introduced by D'Auria et al., 1995 are of great value because they allow testing their capabilities and the plant's response under accident conditions. The elements of the triad method, Ransom et al., 1998, are also part of the Kv-scaled calculation process, as already mentioned.

A Kv-scaled calculation is a system-code simulation in which defined ITF test conditions are scaled-up to an NPP nodalization so to reproduce the same scenario. It allows a comparison of the behaviour of the NPP and the ITF nodalizations under the same conditions. Then, it may be used to check the validity of the NPP nodalization, and to improve it if needed.

The comparison between the results of the NPP nodalization calculations (i.e. the Kv-scaled) results, and ITF experimental data, unavoidably shows differences. The differences are expected, and arise in the comparison between time trends (e.g. pressure, flow-rate, rods' surface temperature, and also pressure drops, level, and the void fraction, etc.). The process of explaining those differences takes into account the scaling factors and adopts the criteria for the qualitative and quantitative accuracy- evaluation criteria, already discussed in previous sections (see also Ambrosini et al., 1990). The process, which may need several months for a single analysis, may end-up – and typically it does – in the discovery of inadequacies and undue approximations in the NPP nodalization, and in some cases errors made by the analysts. Of note are the following:

– A comprehensive description of Kv-scaled analysis requires details which are (only) summarized in the text below.
– Processes different from Kv-scaled calculation can be used to detect inadequacies in NPP nodalization, approximations, and errors: Kv-scaled has proven to be an efficient and traceable process and is strictly connected with scaling (this is the reason why the S-SOAR mentions the synthesis of this method).

In a generic NPP Kv-scaled calculation, experimental conditions and safety actions are adapted without modifying the NPP nodalization. The most significant parameters are the following ones:

– Steady-state conditions
– Break size
– Break unit and containment
– Core-power decay curve (if it is experimentally imposed)
– Pump coast-down curves (if they are experimentally imposed)
– Scram set point
– Isolation set points
– ECCS's set points
– ECCS injection curves (pressure versus mass-flow curves)
– Blow-down set points
– Specifications of the blow-down valves (area, opening- and closing- ratios)
– Feed-water controllers.
– PZR-heater controllers (If this is the case)

A scaling adjustment is performed (this can be a scaling-up adjustment if one starts from the ITF data or a scaling-down adjustment of one starts from NPP data), starting from the original NPP nodalization and by introducing the changes, e.g. in the value of parameters listed above. This adjustment is made by following the scaling criterion and using the scaling factors recalculated for the specific NPP's nodalization. These usually differ from those used in the ITF design (related to the ITF reference plant). An example is provided below (in italics):

Let's consider the reference NPP, and the related nodalization, equipped with four accumulators (design feature of the NPP). The ITF concerned during this experiment may have only one accumulator active in the cold leg, and the pressure of the accumulator may differ from that in the reference NPP. In the NPP Kv-scaled nodalization, only one 'ideal' accumulator is considered as having the same parameters as in the ITF experiment (not a comprehensive list, which should also include thickness of accumulator vessel, for example.):

– The initial pressure
– The L/D for the accumulator vessel
– The L/D, including local pressure drops for the surge-line
– The ratio between accumulator's liquid mass and the coolant mass in the primary circuit.
– The connection point in the cold leg.

In this situation, the elevation of the initial liquid-level related to the top of the active fuel remains different between the ITF- and the Kv-scaled NPP-nodalization, and may be at the origin of differences between experimental data and the NPP Kv-scaled calculation. Those differences can be investigated with a deeper analysis as discussed below.

The presented scaling-up techniques apply to ITF tests performed in facilities that were designed using the Power-to-Volume scaling criterion (the so-called Kv-scaling) which encompasses time-preserving scaling.

One of the important points of this activity lies in calculating the NPP scaling factor (the Kv-factor) that was commonly computed as the ratio between the volume of the primary liquid of the NPP and the ITF. A deeper consideration of this criterion is useful within the Kv-scaled analysis. For instance, some RCS components (e.g. PRZ, SG plenums, pumps, etc.) of the reference NPP may have scaled down values different from those of the concerned ITF, also due to the specific design of the ITF reference NPP, Martinez-Quiroga, 2014. This necessitates an evaluation of the geometric configuration of the concerned ITF, and may require specific additional calculations within the framework of the Kv-scaled analysis.

Subsequently, the analyst should calculate the scaling factor as an average of the same local factors applied to the chosen NPP. Normally, core power, core volume, and the total number of U-tubes (for the PWR) are a good reference.

Hidden or implicit steps in the original kv-scaled methodology have been formalized, applied to relevant samples, and made more traceable within the UPC Scaling-Up Methodology, Martinez-Quiroga & Reventos, 2014, Martinez-Quiroga et al., 2014, and Martinez-Quiroga, 2014. The procedure is a systematic approach for qualifying NPP nodalizations by extrapolating the Integral Test Facility's (ITF's) post-test simulations. The procedure includes defining and using concepts like 'hybrid nodalization' and 'scaled-up nodalization'.

Both concepts were shown to be helpful in justifying the qualitative discrepancies that exist between calculations for the ITF and its Kv-scaled counterpart. 'Scaled-up nodalization' allows the effects of the criterion for ITF scaling-down to be checked. On the other hand, 'hybrid nodalization' helps the user to establish how differences in design modify the results. To carry out these calculations, a Power-to-Volume-Scaling Tool (PVST) was developed. This software generates scaled-up input decks for RELAP5/mod3, following the PVTS approach. In the framework of the NEA ROSA-2 and PKL-2

Counterpart Test, both Methodology and PVST software were assessed, justifying and demonstrating the discrepancies that were reported in the PKL and LSTF experiments (the CET versus PCT correlation, and the delay in core dry-out delay), Martinez-Quiroga, 2014.

4.3 Scalability issues – scaling-up

In the entire process of resolving a transient in the thermal-hydraulic reactor, the system code plays an important role in scaling-up from SETs and IETs that is used for validating the reactor's application. If the code has good "scalability" i.e. when it is can be used to reactor simulate the reactor with the same accuracy as observed in SET- and IET- simulations relative to the same situation. Some limitations of the system code with respect to scaling are listed in section 4.1.4. They are relative to the inherent limits of 0-D, 1-D, 2-D, and 3-D models, to the limits of closure laws, including flow-regime transitions and to non-modeled phenomena. Although there are many sources of possible up-scaling deficiencies in system codes, one should not conclude that the system code's "scalability" is poor since many sources of deficiency may play a role in a bad comparison between experimental data and calculated results.

This up-scaling capability indeed is often reliable as was shown by using system codes to simulate new experiments of a larger scale than the previous ones used for validation. Reflooding models first were validated extensive on several forced-feed experiments in rod bundles of a relatively small size. Then, the validated codes simulated larger scale reflooding tests, such as the PERICLES rectangular tests with 7X51 rods, or the SCTF with 8 full-sized rod bundles. Although additional 3-D effects were present due to the radial-power profile, PERICLES, Rectangular ones were predicted with the same accuracy by the CATHARE code than the previous smaller tests, such as the ERSEC tests, Morel & Boudier, 1999. Later, the SCTF tests also were simulated with reasonable accuracy, with the CATHARE code tests, Dor & Germain, 2011, demonstrating again a reasonable prediction in this very large-scale facility and showing good scalability.

However, some scalability issues also were observed when using system codes for new larger-scale experiments. For example, the Direct Contact Condensation at ECCS injection was modeled in the CATHARE code using reduced scale COSI tests, Janicot & Bestion, 1993, and the model did not generate significant errors when simulating many transients in several integral facilities (e.g. BETHSY, LSTF and LOFT). However, when simulating full-scale (or scale-1) UPTF tests, a significant underestimation of condensation was observed under some high SI-flowrate conditions. The resulting errors at the reactor scale may not be very important on many LOCAs and other transients since it induces a limited error in the pressure history. However, when investigating some Pressurized Thermal Shock scenarios with two-phase conditions at the ECCS location, the required accuracy may be higher than for analysing core cooling during a LOCA. Therefore, the scalability of the DCC may become an issue for PTS. This shows that some scaling-up deficiencies may be accepted in some domains of application of the code but may reveal a scalability issue in applications to new scenarios, including new reactors conditions.

Therefore, it is necessary to list the various scalability issues and to give an evaluation of their impact on predicting reactor transient. Scalability issues relative to closure laws, to code development, to verification and validation, and to code applications are considered below.

4.3.1 Scalability issues related to each closure law

Scalability issues related to closure laws in system codes may originate due to the following reasons:

– A purely empirical correlation or a semi-empirical closure law is applied beyond the domain of experimental investigation. This may occur particularly in large-diameter pipes, such as the hot legs or cold legs of a PWR since there are very few SET data in such diameters. However, some important phenomena were investigated in the UPTF full-scale test facility including a CCFL in the Hot Leg, entrainment in the hot leg, ECCS discharge in the hot leg, and/or in the cold leg.

Although UPTF is not a SET since it does not provide precise boundary conditions for each component; a system code validation on UPTF tests prevents from big issues of scalability in pipes of large diameter.

- Closure laws are based on the assumption of established flow, which is no longer valid in many situations encountered in reactor circuits. Some pipes in cooling loops may have a short length-to-diameter ratio. Transitions in the flow regime in two-phase flow have time- and length-scales that are not taken into account. Here again, the validation of UPTF tests prevents big issues of scalability issues in cooling loops.
- The phenomenological- or mechanistic-closure laws properly account for the scale effects as far as physical assumptions are valid. Due to the complexity of two-phase flow phenomena, the use of such mechanistic laws beyond the domain of validation also may be dangerous. Experience shows that two-phase thermal-hydraulics contain myriads of phenomena that make it difficult to generalize any theoretical model. Interpolation often is possible but extrapolating the scale is dangerous even with physically based models. Having a few large-scale data in a validation matrix may reduce the risks of such closure issues.
- Some mechanical closure laws for wall- and interfacial-friction do not depend on the interfacial heat and mass transfers, although these transfers may have significant effects. The flashing process in a choked flow creates many small bubbles, and direct contact condensation (DCC) collapses all of them. In both cases, classical flow regime maps no longer are valid. Such issues partially are solved by a global validation of all interfacial- and wall-transfer models in flashing flow and condensing flow at different scales, including large-scale ones (Marviken for flashing, and UPTF for DCC).
- All codes model part of the phenomena and ignore or neglect others. System codes neglect many, as was discussed in previous subsections. One may imagine that many other phenomena that were not modeled may have some influence on reactor transients. If these phenomena exist in both IETs and reactors, the system codes may have an estimation of the impact of the phenomenon on the transient. At least, this may be taken into account in the global uncertainty of the code's predictions. If the non-modeled phenomenon plays a larger role at the reactor scale than in scaled IETs, there is a scaling issue. Examples of non-modeled phenomena are the following:
    - Most system codes lack modelling for turbulent diffusion and dispersion modelling in the 2-D and 3-D models.
    - The absence of modelling of axial diffusion and axial dispersion of momentum and energy in most 1-D models (some options to model axial diffusion-dispersion may exist in some system codes) for some applications where it may become a sensitive phenomenon, e.g. boron dilution or mixing of hot- and cold-water in a MSLB.
- Closure laws used in 2-D and 3-D models generally are extrapolated or simply taken from 1-D models. Wall-transfer coefficients do not depend on the direction of the velocity with respect to the fuel rods. Flow-regime maps are the same as in 1-D models although the existence of radial velocity probably may affect the bubbles' size, the droplet entrainment and deposition. Moreover, interfacial friction is isotropic in a non-isotropic medium.

4.3.2 Scalability issues related to the development of codes

The code design introduces assumptions and simplifications that may jeopardize scalability; moreover, code development may include coding errors or numerical errors that affect to some extent the scalability. These limitations are listed above in section 4.1.4.

Most of the limitations are well identified, and can be addressed by the following actions:

- Effects of local complex geometries that have a significant effect are taken into consideration by performing experiments and implementing ad hoc sub-models for form-loss coefficients, CCFL limits, choked flow-multipliers, and specific models for separators, dryers, pumps, turbines, valves, safety valves, control valves, check valves and flow-limiters. Spray cooling and guidelines are provided to users to apply such tools properly. Issues may only remain in the absence of prototypical experimental data.

– Coding errors are tracked by extensive V & V processes, and numerical errors are quantified in the verification process.

Some other limitations remain difficult to quantify in all situations, as stated in section 4.1.4.3:

– The use of O-D or lumped models in some reactor components may become an issue in situations where specific 3-D effects play a dominant role:
  - o Temperature stratification in some components may be destroyed by some local-flow configurations, resulting in a sudden increase of the interfacial heat transfers (sudden high-condensation rate).
  - o O-D modules sometimes are used for multi-connection components with rather high velocities, such as in the upper part of a downcomer connecting all cold legs with the upper head and the downcomer. Predicting the pressure field and pressure losses for all possible flow directions at each connection is far beyond the capabilities of such lumped models.

– As stated in section 4.1.4.4, using a 1-D model also may also become problematic in some situations. One strong assumption is that all the transfer terms can be expressed as function of local principal variables only. These functions often are algebraic functions and may also be functions of local derivatives of the principal variables (e.g. added mass force, the interfacial pressure gradient, or 'Pi grad', the term). Limitations are associated with this strong assumption:
  - o There are situations where the velocity field is not exactly unidirectional, in particular, when the effects of natural convection play a role.
  - o Closure laws cannot be functions of the abscissa along the flow in the duct, and cannot take into account the establishment of the flow: wall-friction and heat-transfer coefficients consider established flow conditions.
  - o History effects on the turbulence level and on the two-phase flow regime cannot be taken into account. It is very limitative to model an average bubble size in bubbly flow or an average drop size in annular-mist-flow. Coalescence, break-up, nucleation, collapse, vaporization, and condensation may affect the size of the bubbles or drops. Algebraic models for their size cannot account for this dynamic behaviour and can only be tuned on some data. However, differences may exist between experiments and reactors with respect to these effects and, at least, these closure laws are not scalable. In particular, the behaviour of entrained droplets between core and steam generator during a reflooding process cannot be scaled properly by current system codes.

The scaling issues mentioned here above cannot be solved unless validated 3-D modules replace the 0-D modules or transport of interfacial area is provided in the modelling; furthermore, validated closure laws shall be developed to predict the dynamic behaviour of droplets and the bubble's size. Before such advanced models become available these issues only can be addressed within the framework of the evaluation of uncertainty.

4.3.3 Scalability issues related to verification of the code

As shown in section 4.1.2, the process of verifying the code checks the implementation of the numerical scheme and measures its accuracy. The objective is to detect coding errors, and to control numerical errors related to the numerical scheme.

Coding errors may induce any type of error and they may include those ones that affect the code's scalability. Consider a coding error that would be quantitatively small in reduced-scale test simulation, and which would have a higher impact at reactor scale.

Also, one may imagine numerical errors that may have a greater effect at the reactor's scale than at a reduced scale. This seems possible since no system code writes equations in non-dimensional form.

The way to detect scalability issues due to either coding or numerical errors is to undertake a few control tests at different scales and to measure their scalability.

Such "numerical scalability tests" usually are not performed in the current verification processes but may be added.

4.3.4 Scalability issues related to validation of the code.

To verify the quality of the physical modelling, system codes are extensively validated. The validation process occurs in several steps:

- Comparison of model results with the results of basic tests and/or separate effects tests (Limitations: The basic and/or separate effect tests (SETs) mainly are performed in down-scaled test facilities or in a scaled reactor-component).
- Comparison of code results with the results of integral system tests (IETs). (Limitations: all integral system tests are performed in scaled-down test facilities).
- Comparison of code results with plant data of transients and accidents. (Limitations: data are only available for a very limited domain for the range of conditions of variables which characterize transients – mainly start-up ones – and accidents).
- Comparison of results calculated with different models or codes (benchmark calculations) (Limitations: Agreement between the results does not signify necessarily agreement with the real physical process that has been simulated).

Taking into account the limitations listed above, a clearly structured, goal-oriented test-and-validation strategy must be developed for the different types of codes. The CSNI validation matrices, NEA/CSNI, 1987, NEA/CSNI, 1993, NEA/CSNI, 1996, and NEA/CSNI, 2001, are the key elements for such a strategy.

Since a full-scale integral test is impossible, the scaling of the test facilities becomes a fundamental issue. The question of scaling has to be considered not only in the design of the test facilities; it has also to be taken into account in the process of code development.

The codes must calculate satisfactorily the results from separate effect tests, and from integral or systems effect tests at different scales (for example, volume scaling of integral system tests range from 1:1 to 1:1600). For the full range of variables that characterize transients and accidents, the interactions of the code models can be checked only on the results of down-scaled integral-system test facilities. On the other hand, code models developed mainly based on the results of down-scaled experiments must describe the thermal-hydraulic processes expected in the full-scale reactor plant (scale-up capability).

4.3.4.1 Issues related to validation of the code by SETF data

SETs are the best way to validate each physical model or closure law, and each phenomenon specific to a particular geometry of a reactor component. When closure laws have good scaling properties, they can be used to extrapolate the accuracy from a reduced scale test to a reactor condition. However, due to the complexity of the two-phase flow, there is no guaranty that the specific model takes into account all the important flow-processes that may have an influence and that all the relative important non-dimensional numbers are present in the model. Difficulties may come from a process that has a low impact at a small scale, and a higher effect at higher scale. If the model is validated only at a small scale, one cannot see that an effect is not described. The inverse situation may exist: scale dependence may be observed at small scale that no longer exists at larger scale.

For example, the slug flow in a vertical pipe introduces a clear dependence of the diameter in a drift-velocity model, or a corresponding interfacial friction model. However, large Taylor bubbles occupying almost the whole diameter of a pipe no longer exist for pipes larger than about 30 times the Laplace scale, and the so the diameter dependence disappears for them. First versions of system codes had not identified well this change in diameter dependence, but validation on larger and larger sets of pipe diameters led to more complex drift flux models, and interfacial friction models that have resolved the scalability issue. For example, the Cathare model, Bestion, 1990, or the Kataoka & Ishii, 1987, model, or even the Chexal-Lellouche models, i.e. Chexal & Lellouche, 1991, and Chexal et al., 1992, now have reasonably good scale-up properties for this phenomenon.

In addition to the integral-system tests, separate effects tests are being performed for investigating particular phenomena. There are several reasons for the importance and high value of this type of test.

Firstly, it was recognized that the development of individual code models often requires some iteration, and that a model, despite its being well-conceived, may need refinement as the range of applications widens. To establish a firm need for modifying or further developing a model it usually is necessary to compare predictions with data from separate-effects tests, rather than to rely on inferences from integral-test comparisons.

Secondly, a key issue concerning the application of best estimate codes to calculating LOCAs and transients is the quantification of uncertainties in predicting safety-relevant parameters. Most methods for determining these uncertainties rely on assigning uncertainties to the modelling of individual phenomena. This concept placed a new emphasis on separate-effects tests above that originally envisaged for the model's development.

The advantages of separate-effects tests (over integral effect tests) are the following:

- Clear boundary conditions;
- Measurement instrumentation can be focused on a particular phenomenon;
- Reduced possibility of compensating errors in modelling during validation;
- More systematic evaluation of the accuracy of a code model across a wide range of conditions, up to the scale of a full-reactor plant;
- Steady state- and transient-observations are possible.

A further incentive to conduct separate-effects tests in addition to experiments on integral- system test facilities is the difficulty encountered in the up-scaling of predictions of phenomena from down-scaled integral tests to real plant applications.

Where a phenomenon is known to be highly scale-dependent and difficult to model mechanistically, there is a strong need for conducting separate effects tests at full scale. Examples are given here where only scale-1 tests could solve properly the scaling issue.

Mainly selected results of the 1:1 scaled Upper Plenum Test Facility (UPTF) are shown in the following sections. The chapter also contains new scaling laws developed for designing separate- effects test facilities for investigating the direct-vessel ECC bypass phenomenon in the downcomer of power reactors with direct ECC injection into the downcomer, instead of into the cold legs.

UPTF in Germany was a full scale (1:1) representation of the primary system of a 1300 MWe pressurized water reactor with four-loop system, Weiss et al., 1986, Fig. 4-5. It is a fluid-dynamic facility, especially designed for studying the effects of multi-dimensional two-phase flow in components of large volume, like the downcomer, upper plenum, entrance region of the steam generator, and the main coolant pipes. The steam generation in the core and the entrained water flow during emergency cooling was simulated by feeding steam and water via nozzles in the so-called core simulator. The behaviour of the steam generators was replicated by a controlled removal or feed of steam and water from or to the steam generator simulator from outside. The required steam was provided by a fossil-fuel-fired power station.

Fig. 4-6 provides a picture of the core simulator and all the pressure-vessel's internals.

The data from the UPTF have remarkably extended the data base required to develop and validate analytical models used in the extensive thermal-hydraulic codes for simulating two-phase flow phenomena in the geometry of a full-scale reactor. The large-scale test facility has shown that particular multidimensional phenomena cannot be simulated in test facilities at smaller scales, e.g. Glaeser & Karwat, 1993.

The results of these multidimensional flows in the downcomer, in the upper plenum, and at the upper core tie plate, as well as those related to flow phenomena in the hot and cold legs of the main coolant pipes are given below.

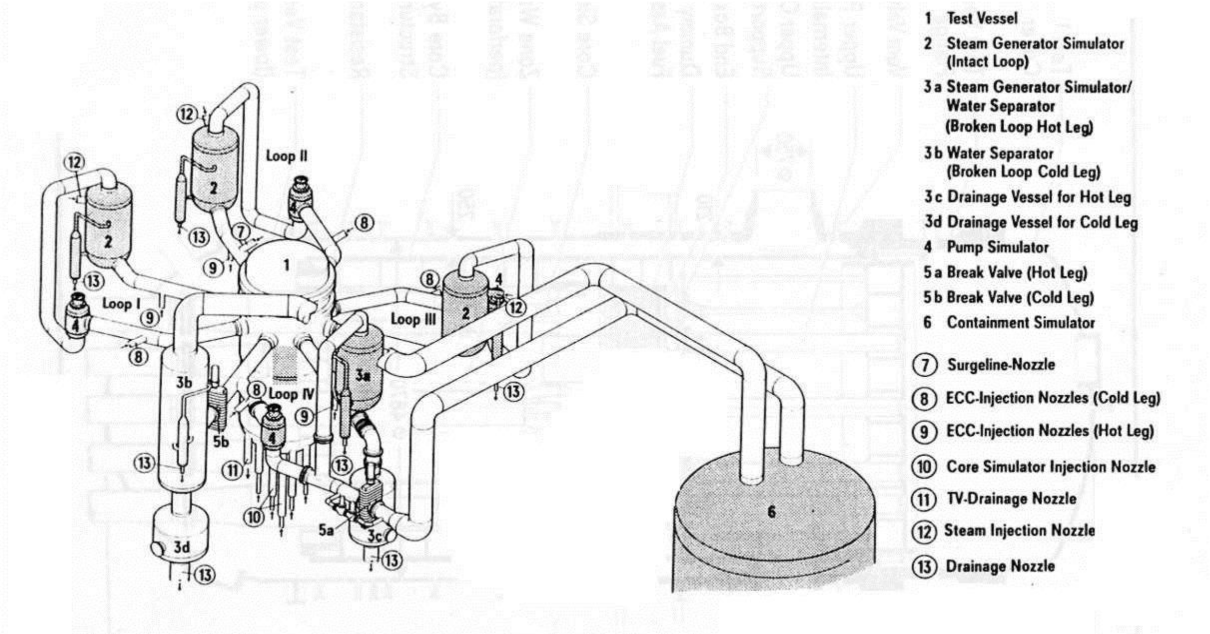


Fig. 4-5 – UPTF, the German contribution to the trilateral 2D/3-D Program.

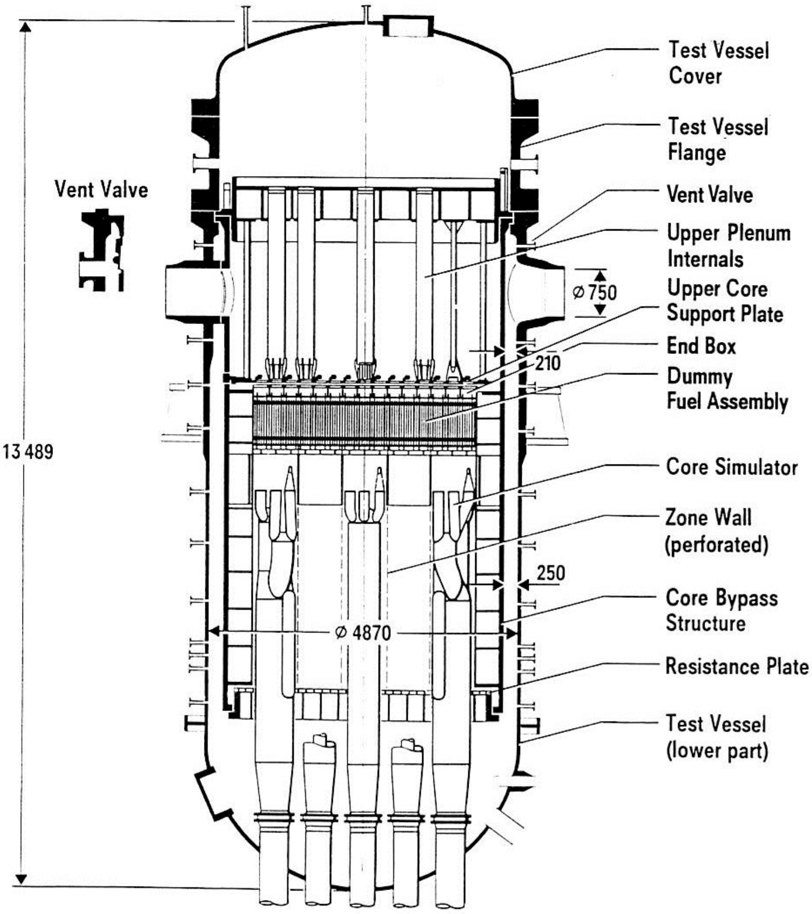


Fig. 4-6 – Section of vessel of the Upper Plenum Test Facility (UPTF).

Thermal-hydraulic phenomena in the downcomer

Previous insights into downcomer phenomena during cold leg ECC-injection were developed particularly based on the USNRC ECC Bypass Program. It contained steam-water tests which were performed at 1/30, 1/15, 2/15, and 1/5 scale at the Battelle Columbus Laboratories, and at CREARE, e.g. Crowley et al., 1977. Steady-state counter-current flow limitation (CCFL) tests with steam up-flow and ECC-water down-flow were carried out, as well as transient tests involving flashing in the lower plenum and two-phase up-flow.

Empirical flooding correlations based on the experimental data from these down-scaled test facilities were developed, using two dimensionless phase velocities. A modified Wallis parameter containing 'W' as the average circumference of the downcomer annulus, was proposed,

$$J_k^* = \frac{\dot{M}_k}{\rho_k A_{DC}} \frac{\rho_k^{1/2}}{[gW(\rho_w - \rho_s)]^{1/2}} \tag{4-5}$$

Together with a specific and the Kutateladze number based correlation,

$$K_k^* = \frac{\dot{M}_k}{\rho_k A_{DC}} \frac{\rho_k^{1/2}}{[g\sigma(\rho_w - \rho_s)]^{1/4}} \tag{4-6}$$

Despite the variation of scale of these test facilities between 1/30 and 1/5, the extrapolation to full scale downcomer geometry still was not clear.

To provide data on the CCFL and by-pass for full-reactor geometry, tests on UPTF were carried out.

Figure 4-7 shows the results of a UPTF experiment, simulating the downcomer behaviour during the end-of-blowdown and the refill phases for a large cold-leg break.

The test was carried out at a reactor typical steam up-flow of 320 kg/s, and ECC-injection (sub-cooling 115 K) into the three intact loops. The contour plot shows the isotherms of fluid temperatures (sub-cooling) in the downcomer, projected in a plane. The two-dimensional representation shows strongly heterogeneous flow conditions that were not observed in small-scale experiments. The ECC-water delivered from the cold legs 2 and 3, which are located opposite the broken loop, penetrates the downcomer without being strongly affected by the up-flowing steam. However, most of the ECC-water delivered from cold leg 1, which is located close to the broken loop, flows directly to the break by-passing the core.

To demonstrate the effect of the facility scaling on the downcomer CCFL, the data obtained from UPTF and CREARE with 1/5 scaling are compared in Fig. 4-8, showing the Wallis parameter as defined before. To compare data of slightly subcooled ECC water from UPTF with the CCFL results of CREARE obtained with saturated ECC-injection, an effective flow of steam (injected minus condensed steam) was introduced.

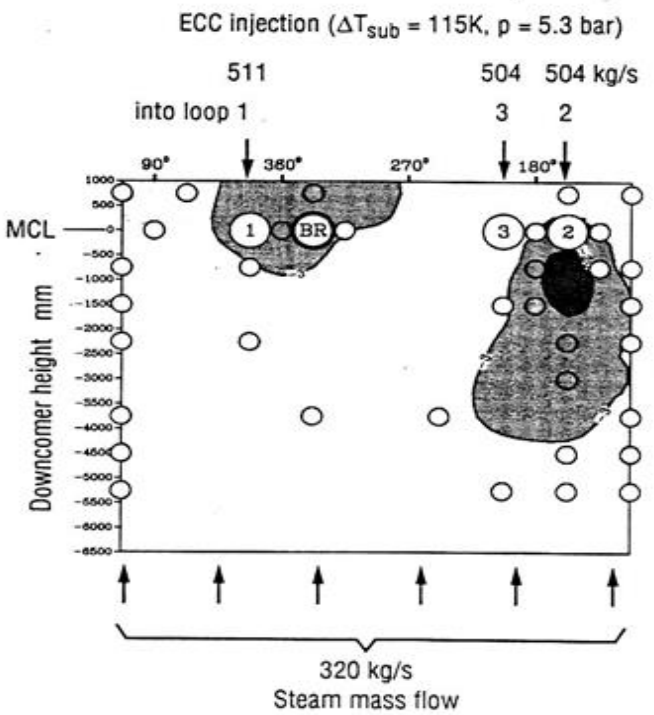


Fig. 4-7 - Counter–current flow conditions in full-scale downcomer for strongly subcooled ECC – distribution of sub-cooling temperatures.

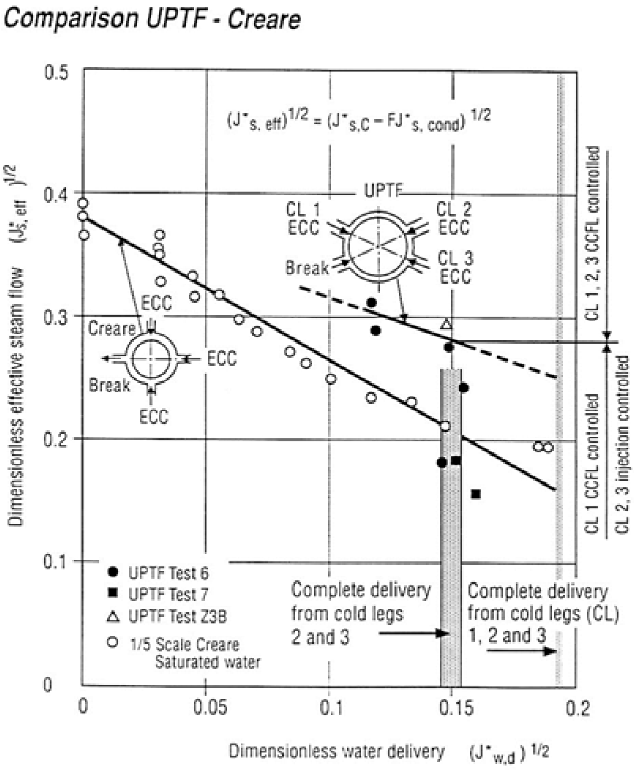


Fig. 4-8 – Effect of loop arrangement on water delivery to lower plenum for nearly saturated ECC water in counter-current flow.

Due to the strongly heterogeneous flow conditions in the full-scale downcomer of the UPTF, the water delivery curves of UPTF and CREARE are significantly different. For dimensionless effective steam flow, $(J^*_{s,eff})^{1/2}$, greater than 0.2, the dimensionless water down-flows of the UPTF are much higher than the results of CREARE. We note that the UPTF data at dimensionless effective steam-flows smaller than 0.2 below the CCFL curve should not be directly compared with the CREARE CCFL curve. These data points are limited due to the ECC injection rates into cold legs 2 and 3. Higher rates of water delivery could be expected if more ECC-water were injected into these two cold legs.

The main findings with respect to downcomer behaviour during the end-of-blowdown and refill phases of a large cold leg break with cold leg, or downcomer ECC-injection can be summarized as follows:

- A significant scale effect on the downcomer’s behaviour can be observed,
- Flow conditions in the downcomer are highly heterogeneous at full scale,
- Heterogeneous- or multi-dimensional-behaviour increases the delivery rates of liquid into the core at full-scale compared to what measured in previous tests on down-scaled facilities,
- CCFL correlations developed from the down-scaled tests are not applicable to full-scale downcomers,
- Downcomer CCFL correlations for cold-leg ECC injection based on the results of down-scaled tests underestimate the extent of water penetration into the lower plenum at full scale,
- Due to strong heterogeneous flows in a real downcomer, a CCFL correlation must account for the location of the ECC injection relative to the break.

To describe the vertical asymmetric heterogeneous gas/liquid counter-current flow in a full- scale downcomer, the Kutateladze type flooding equation was extended by Glaeser, 1989, correlating the local steam velocities of the multi-dimensional flow field with the superficial steams' velocity. A geometrical lateral distance between the legs with ECC injection and the broken loop is introduced in the gas up-flow momentum term. This term relates the local upward velocity of gas at the water down-flow locations to the superficial gas velocities. The latter can be calculated from the steam's mass flow rate:

$$K_g^{1/2}\left(\frac{v_g^{2/3}}{g^{1/3}L}\right)^{1/2} + 0.011K_l^{1/2} = 0.0245 \tag{4-7}$$

$$\frac{g^{1/3}L}{v_g^{2/3}} \geq 5400 \qquad\qquad L = 0.5\pi\, d_{outer} \sin^2\left(0.5\Theta_{ECC-BCL}\right)$$

If there is more than one location of ECC injection, the arithmetic mean value of all distances, L, between the ECC injection legs and the broken leg have to be used in the correlation. However, only those injection locations can be considered where water can flow downwards. This means that the modified dimensional gas-velocity obtained by using the value of L for the individual injection location must be below the onset of penetration point. Otherwise, the respective ECC injection leg cannot be included in the arithmetic mean value, L. More details of the derivation and application of the Glaeser-correlation can be found in Glaeser, 1992, and Glaeser & Karwat, 1993.

The resulting lowest gas velocity for zero water penetration (the onset of penetration) is shown in Fig. 4-9, compared with the scale of the downcomer's circumference.

The down-flow of water is impossible for gas velocities above the given curve. There are three different scaling regions for which one of the flooding correlations is applicable. These are the classical Wallis and Kutateladze types, as well as the Glaeser correlation. The range of applicability depends upon the dimensionless circumference of the annulus that governs the different flooding correlations. It is evident that it is impossible to extrapolate counter-current flow correlations from small-scale data below one-ninth of the downcomer's circumference scale (equivalent to a 1/81 flow cross-section scale) to the reactor's scale. The full-scale UPTF data were needed to clarify the influence of scaling on the ECC's flooding phenomenon.

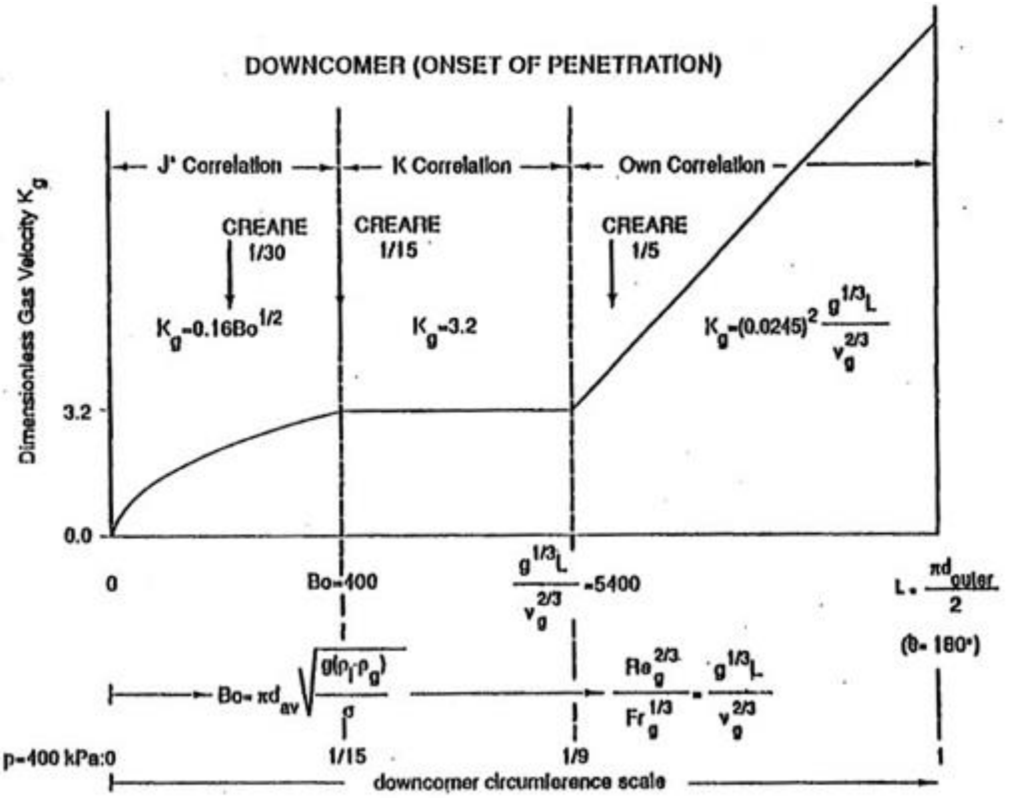


Fig. 4-9 – Downcomer flooding correlation for zero penetration of liquid (total bypass).

Thermal-hydraulic phenomena at the tie plate, and in the upper plenum

Dependent on the type of ECC-injection systems, different flow-phenomena occur at the tie plate and in the upper plenum of a PWR. For PWRs with cold-leg injection or downcomer ECC injection, a counter-current flow of steam/water up-flow, and saturated water down-flow occurs. The water, which is entrained by the up-flowing core-steam flow, either is de-entrained at the tie plate, de-entrained in the upper plenum, or carried over to the hot legs. The saturated water that is de-entrained in the upper plenum either forms a pool in the upper plenum, or flows counter-currently to the steam/water up-flow back through the tie plate and into the core. For PWRs with ECC injection into the hot leg, or the upper plenum, counter-current flow phenomena at the tie plate involve steam/water up-flow and local down-flow of subcooled water.

Before the UPTF tests were performed, the knowledge about the thermal-hydraulic phenomena at the tie plate and in the upper plenum was based on results gained from scaled-down test facilities. The tie plate usually was simulated by small perforated plates not exceeding the size of one fuel assembly. The Wallis parameter or the Kutateladze number was applied to correlate the experimental data. To study the behaviour of the tie plate and the upper plenum in full reactor geometry, tests on UPTF were undertaken. They were carried out with three different types of thermal-hydraulic boundary conditions, Fig. 4-10:

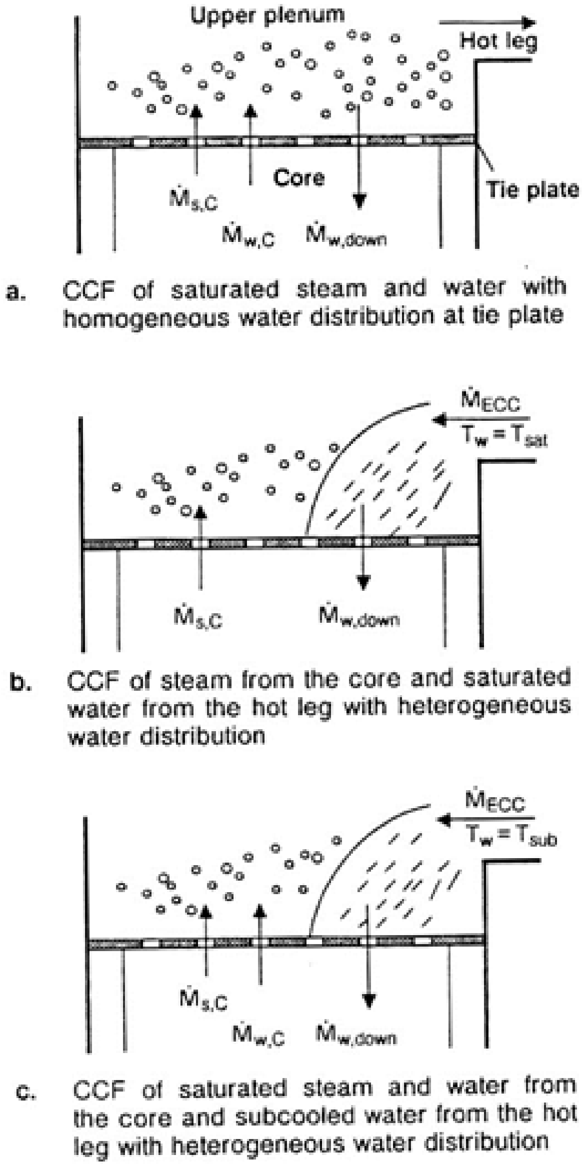


Fig. 4-10 – Counter-current flow conditions at the tie plate addressed in UPTF.

1) Counter-current flow of saturated steam and water at the tie plate, typical for PWRs with cold leg ECC injection,

2) Counter-current flow of steam and saturated water injected into the hot legs,
3) Counter-current flow of saturated steam and water from the core and subcooled water injected into the hot legs, typical for PWRs with combined ECC injection.

To study the counter-current flow at the tie plate and the liquid hold-up above the tie plate in case of saturated steam/water up-flow, a series of UPTF tests were carried out. The reactor's typical steam/water up-flow was adjusted by the core simulator via a controlled injection of steam and water. The upward flow of steam and entrained water is of higher relevance for counter-current flow in the reactor than the steam up-flow alone. This effect had not been investigated in tests of vertical counter-current separate effects prior to the UPTF and UPTF calibration tests at the Karlstein facility consisting of one single fuel-assembly.

In Fig. 4-11, data from these UPTF tests are plotted using the Kutateladze number for up-flowing steam, and down-flowing water. In addition, corresponding data are presented from single fuel-assembly tests performed at the Karlstein Calibration Test Facility to determine potential scale effects. The figure clearly shows that a limiting counter-current flow at the tie plate is occurring at the same Kutateladze number in the single-fuel assembly test facility as in the full-scale sized facility, UPTF, with a cross section of about 20 m^2.

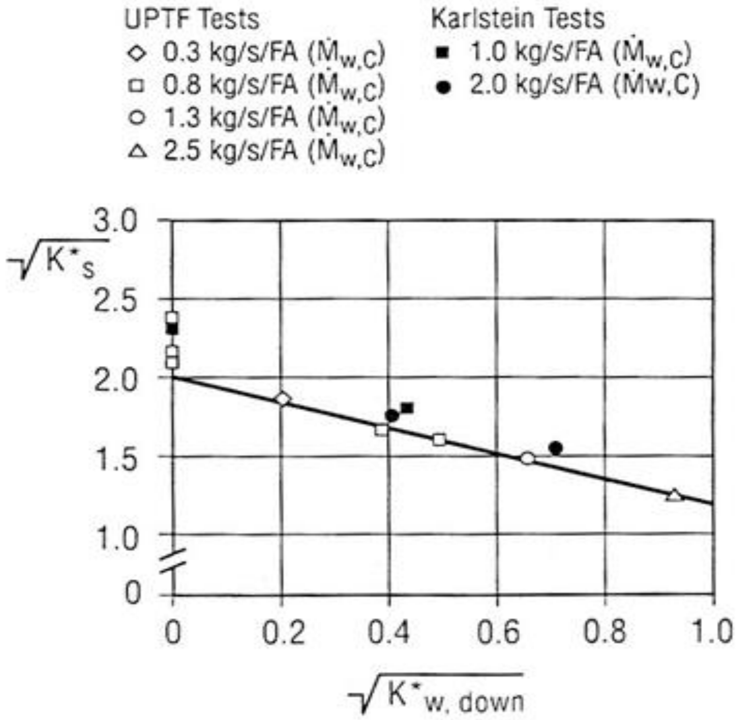


Fig. 4-11 – Counter-current flow of saturated steam and water at the tie plate for homogeneous-flow conditions.

The test results indicate the following:

- Steam/water up-flow, the two-phase pool above the tie plate, and water fallback through the tie plate is uniform across the core area,
- Flooding curves for both the full-scale and sub-scale test facilities are similar,
- Water down-flow to each fuel assembly is scale-invariant,

For conditions of homogeneous flow at the tie plate, the flooding curve can be defined by applying the Kutateladze number as scaling parameter.

The situation for countercurrent flow of steam and saturated water injected into the hot legs differs strongly from the one described above in that saturated ECC water is delivered to the upper plenum via the hot legs, while steam is injected through the core simulator flowing upward through the tie plate. This boundary condition is not typical in reactors; however, tests with saturated-water injection allow the investigation of heterogeneous flow distribution in the upper plenum and the tie plate region without the influence of condensation effects.

A series of UPTF tests were carried out investigating different configurations of the ECC injection. In Fig. 4-12, the results of tests with injection via two loops (injection rates 2 x 100 kg/s) and single-loop injection (injection rate 1 x 400 kg/s) are shown.

The main findings of the tests performed to investigate counter-current flow of steam and saturated water injected into hot loops are summarized as follows:

– Water breakthrough from the upper plenum to the core occurred in front of the injecting hot leg nozzles, leading to heterogeneous flow conditions at the tie plate,
– The paths of down-flow of water and the up-flow paths of steam at the tie plate are separated,
– No substantial time delay occurs between start of ECC-injection and the breakthrough of the tie-plate’s liquid level,
– •The rate of the water’s breakthrough increases with decreasing rate of flow of the core’s steam,
– Non-uniform distribution of vertical differential pressure in the upper plenum is measured,
– Water down-flow is significantly higher than that according to the flooding curve determined from homogeneous flow conditions at the tie plate,
– UPTF tests indicate clearly that classical Kutateladze-scaling cannot be applied for heterogeneous flow conditions without modifications, Glaeser, 1992, and Glaeser & Karwat, 1993.

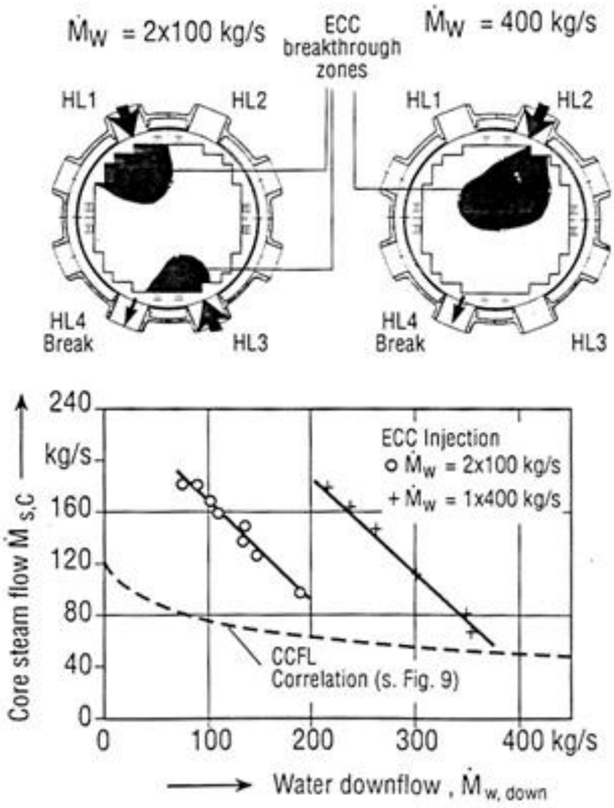


Fig. 4-12 – Counter-current flow of steam and saturated water injected into hot leg.

Compared to saturated hot-leg injection, the conditions for water breakthrough at the tie plate become more favorable if highly subcooled ECC water is injected.

The results of an UPTF test with a very high water/steam ratio of the up-flow rate of w/s=4 are shown in Fig. 4-13 (a typical value for the reflood period of a PWR is w/s=2). Additional results of tests, investigating the effect of the ratio of water/steam of the two-phase up-flow, as well as the effect of hysteresis-flow changes with increasing and decreasing up-flow rates are also presented.

The UPTF tests have shown the following:

– ECC penetration to the core region always follows, without substantial delay, the ECC delivery to the upper plenum, and occurs in front of the hot legs with ECC injection,
– Time-averaged water breakthrough at the tie plate is not significantly affected by intermittent water delivery to the upper plenum compared to continuous delivery,
– Water breakthrough at the tie plate increases with a decreasing flow rate of steam,
– For a given rate of steam up-flow, the water breakthrough at the tie plate increases with decreasing water/steam ratio of the two-phase up-flow,
– The existence of a two-phase pool of saturated water in the upper plenum at the initiation of hot-leg ECC injection has only a minor effect on the water breakthrough at the tie plate,
– During the period of increasing core up-flow rates, the water down-flow is higher than for decreasing up-flow rates at the same steam up-flow rates,
– Heterogeneous flow conditions at the tie plate are strongly dependent on scale; therefore, the classical Kutateladze scaling cannot be applied without modifications, Glaeser, 1992, and Glaeser & Karwat, 1993.

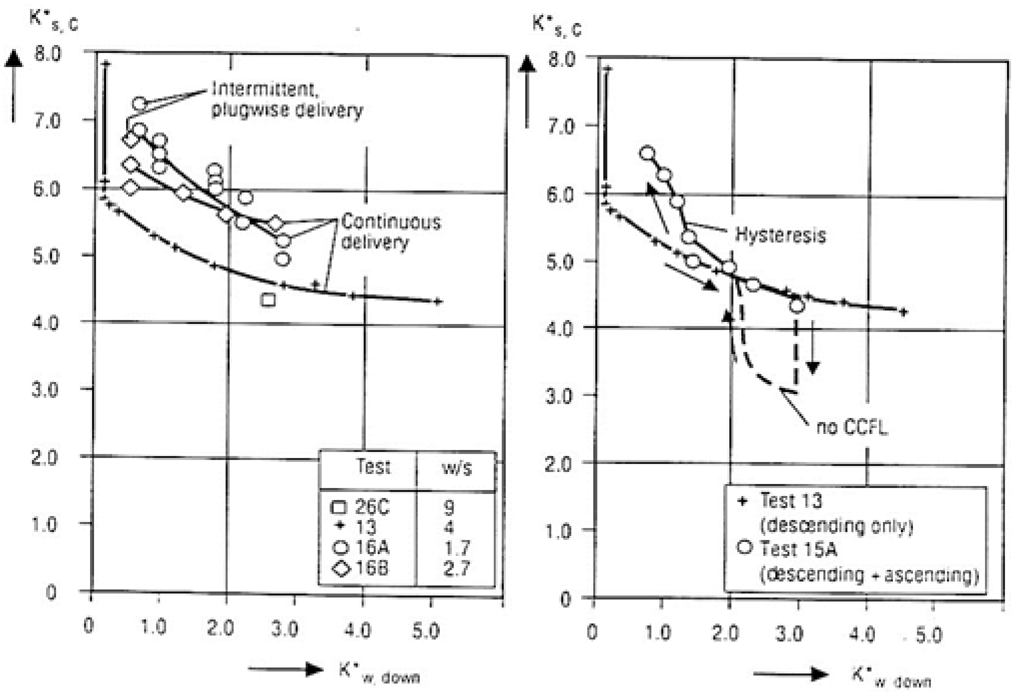


Fig. 4-13 – Counter-current flow of two-phase up-flow and subcooled water down-flow during hot leg ECC injection.

In general, the UPTF tests revealed that the tie plate's CCF behaviour with hot leg ECC injection is quite different from that without this injection, even if saturated ECC-water is delivered to the upper plenum.

The classical Kutateladze-type CCFL correlations can only be used to predict the tie plate water down-flow rate if no ECC water is injected into hot legs or upper plenum, i.e. vertical homogeneous steam-water counter-current flow. Only in this case the tie plate results elaborated in small scale test facilities can be applied to a large tie plate.

In case of hot leg ECC-injection, the water down-flow through the tie plate is significantly higher than predicted by the previous tie plate correlations that are based on small-scale test data. The reason for this deviating CCF behaviour is the vertical inhomogeneous distribution of the water mass across a full size tie plate, due to local ECC water delivery to the upper plenum. Over the full range of the reactor core outlet's flow rates, the injected ECC water penetrates through the tie plate into the core without delay.

The flooding correlations in a hot- leg ECC water injection can be expressed similarly to the correlation for the downcomer's counter-current flow, taking into account the distance between the location of the injection and the broken loop's hot leg. The effects also can be included of the additional upwards momentum by water droplets entrained in an upwards flowing steam, as well as condensation at subcooled ECC water, Glaeser, 1992, and Glaeser & Karwat, 1993.

4.3.4.2 Issues related to validation of the code by ITF data

It is impossible for a down-scaled facility to reproduce all the physical phenomena that occur during a transient process in a real scale plant. The designer must optimize the down-scaled facility for the processes of greatest interest. However, this inevitably leads to distortions of other processes of lesser importance.

The impacts of scaling the test facility scaling are illustrated by the results obtained at the SEMISCALE test facility that was designed primarily to simulate the double-ended cold-leg breaks in a PWR. The design was based on a volume scaling of 1:1600, while retaining of the original plant's elevations, resulting, consequently, in a significant reduction of the flow cross-sections.

Fig. 4-14 compares the cross-sections of the reactor core and the downcomer of a 1300 MWe PWR with the corresponding cross-sections, scaled down by a factor of 1:1600. The core diameter of 400 cm is reduced to one of 5 cm (marked by the red solid circle in the core region). The annular shaped downcomer of a cross sectional area of 35 000 cm^2 is reduced to an area of about 20 cm^2 (marked by the red solid circle in the downcomer region).

Tests performed in the SEMISCALE facility to investigate the effectiveness of Emergency Core Coolant (ECC)-injection in the cold leg and the hot legs demonstrated very clearly the limitations of such small-volume scaling, viz., 1: 1600.

In contrast to the results of the UPTF test facility with a geometrical scaling of 1: 1, the SEMISCALE test results revealed that the penetration of ECC-water into the downcomer during cold-leg ECC injection was prevented completely by limiting the counter–current flow. The multi-dimensional flow in the annulus downcomer of a large PWR could not be reproduced by the small tubular downcomer of the SEMISCALE facility. No effect of cold-leg ECC-injection on core cooling has been observed, which is misleading with respect to large-scale behaviour.

Also, the tests simulating hot-leg ECC injection gave results that which are in full contrast to the UPTF results. Contrary to the break-through of the injected ECC-water via the upper tie plate in the core as is expected in the original plant, and what was observed in the UPTF-tests, was the accumulation of ECC-water in the upper plenum; it did not contribute to core cooling. Even worse, the unrealistic accumulation of the ECC-water in the upper plenum blocked the up-flow of the mixed steam-water coming from the core. These two effects, that definitely are a result of scaling the facility, led to a completely unrealistic picture of the heat up of the core.

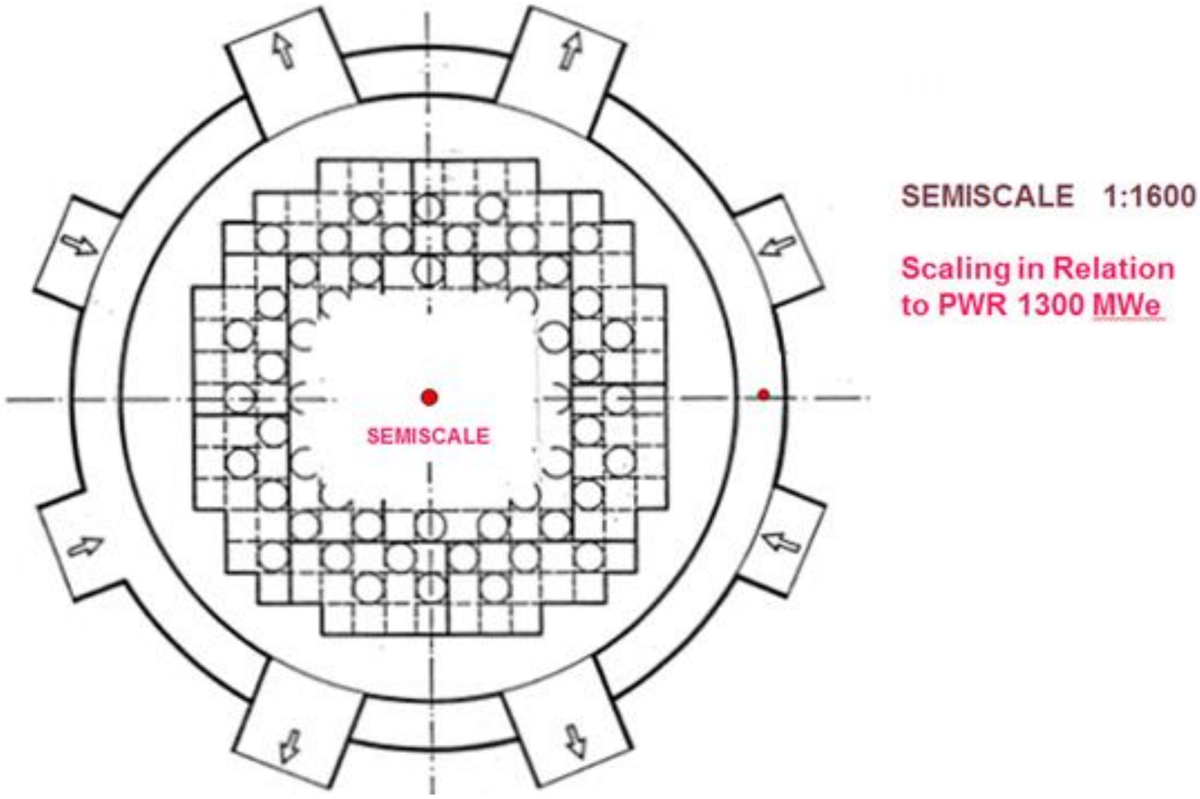


Fig. 4-14 – Reduction of cross sections by the SEMISCALE volume-scaling factor 1:1600.

The influence of scaling components was demonstrated very impressively by the LOBI Program, Riebold et al., 1984. The tests were performed with two different gap-widths of the annular-shaped downcomer. In the first series, a downcomer with a gap width of 50 mm was used; this resulted in downcomer volume that was 6.3 times too large, therefore strongly distorting the mass distribution in the scaled system. The intention was to preserve, as far as possible, the counter-current flow as well as the hot-wall-related phenomena during the refill period. In the second test series, a downcomer of 12 mm-wide gap was installed. The 12-mm was chosen as a compromise between the volume-scaled downcomer (7-mm gap width), and a downcomer that would yield the same pressure drop due to wall friction, as in the reference reactor (25 mm gap-width for the scaled facility). The influence of the width of the downcomer gap and downcomer's volume was shown by comparing the results. Both tests simulated a double-ended cold-leg break with cold-leg accumulator injection. Overall, the initial- and boundary-conditions for the two tests generally were equal or directly comparable.

No significant influence was evident of the width of the downcomer gap on the system's thermal-hydraulic behaviour during the very first blowdown period when subcooled fluid-conditions persisted in the downcomer region. However, the course of the transient strongly was affected during the subsequent saturated blowdown and refill periods. When the fluid also had started to evaporate in the cold regions of the system due to the rate of depressurization, the relatively higher-density fluid persisting near the core's entrance and the re-establishment of positive mass flow through the core were much more pronounced in the case of the large downcomer where the initial liquid inventory in the downcomer is about 3.6 times larger than in the case of the small downcomer. This, in turn, ensured enhanced cooling of the heater-rod bundle during the late blowdown. Consequently, completely different conditions existed in the primary system at the time when ECC-injection from the accumulator started. Conversely, the smaller width of the downcomer tended to inhibit the penetration of ECC water and, hence, lower refilling of the plenum which led to near stagnation conditions in the core and its relatively poor cooling.

These examples show very clearly that the thermal-hydraulic behaviour of a down-scaled integral test facility cannot be extrapolated directly to obtain a picture of the nuclear plant's behaviour. Although the facilities were designed with different volume scaling, ranging from 1:1600 to 1:48, the experimental results from these facilities are not resolving the problems in scaling. A combination of integral system tests with separate effects tests in full-reactor scale thus is indispensable.

4.3.4.3 Issues related to scaling distortions

Using scaling laws for the IET may be problematic for some components, and may result in uncontrolled distortions. One example is the region of a PWR loop that is neither vertical nor horizontal between the Hot Leg and the SG tubes, i.e. the bend and the inlet header of the SG.

Many IETs use a Power-to-Volume scaling associated with a full height, so resulting in difficulties in scaling the bend and inlet SG header. Several experimental facilities adopted various types of distortions. The impact of water retention in some accidental sequences may be an issue, since all the water kept out of the Pressure Vessel is not available to cool the core. Since there is no possibility of preserving the shape of this region of the loop, one may investigate the distortions by SETs and check the code's capability to predict the phenomenon of interest at various shapes and scales. The example below illustrates how the scaling distortion related to flow conditions in the hot leg during reflux condenser mode was solved.

Flow conditions in the hot leg during the reflux condenser mode

In the reflux condenser mode, heat is transferred from the core to the secondary side of the steam generators by the evaporation of water in the core and the subsequent condensation of the steam in the U-tubes of the steam generators. A portion of the condensate flows back counter-currently to the steam through the hot leg via the upper plenum into the core. Due to momentum exchange between the up-flowing steam and the down-flowing water in the hot legs, flooding may occur, which could prevent or at least degrade the water flow back to the core.

Counter-current flow in the hot legs of a PWR was investigated in different sub-scale facilities with pipe diameters up to 200 mm, Fig. 4-15.

To provide CCFL data for full-size geometry, UPTF tests were performed. The results are plotted in Fig. 4-16 using the Wallis parameter J*. They show that water runback to the test vessel declines as the steam flow increases. At high steam flows $J^*s > 0.47$ (i.e. $J^*s^{1/2} > 0.69$), there was a complete turn-around of the flow. The close agreement of the data at the two different measured pressures indicates that the Wallis parameter adequately accounts for the pressure effects.

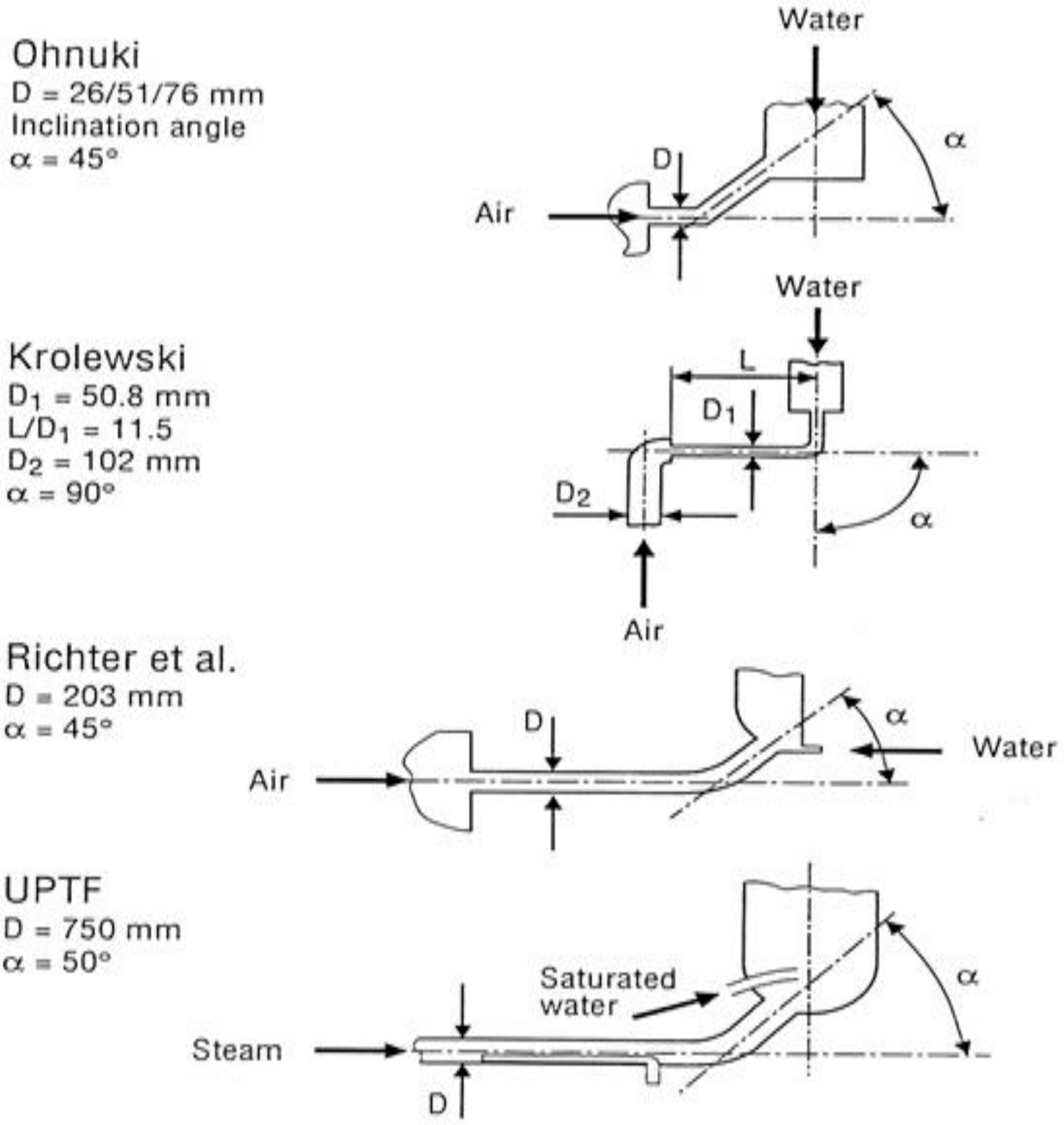


Fig. 4-15 – Counter-current flow of two-phase up-flow and subcooled water down-flow during hot leg ECC injection.

In Fig. 4-16, the UPTF tests are compared to CCFL correlations derived from sub-scale experiments. The Krolewski correlation under-predicts the UPTF water runback; on the other hand, the Ohnuki correlation over-predicts it. The Richter correlation, however, passes through the UPTF data, which undoubtedly is due to the similar configuration of the flow channel.

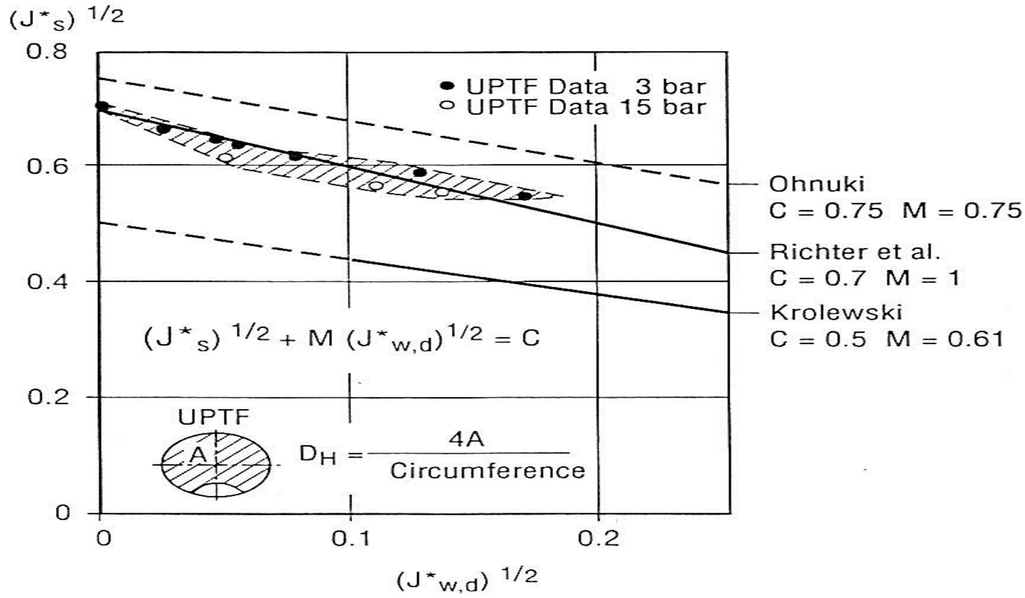


Fig. 4-16 – UPTF test data compared to correlations derived from sub-scaled tests.

This agreement also is an indication that the Wallis parameter is suitable for scaling for mainly horizontal heterogeneous gas-liquid flow. This is the reason why Froude-scaling often is used for scaling the hot-leg dimensions of experimental facilities. The Wallis number or dimensionless velocity J* is equivalent to a Froude number modified by the density ratio of the steam- and liquid-phase.

The Kutateladze-type equation considering the instabilities of the gas-liquid interface in horizontal counter-current flow conditions can be transferred into the Wallis-type equation, Glaeser, 1992. This is possible for horizontal- or inclined-flow since the gravity force of unstable waves and droplets act perpendicularly to the main flow directions of steam and water, and counter the pressure difference between the bottom of such a wave and its crest due to their different velocities, Glaeser & Karwat 1993. Hence, the Wallis correlation is applicable to horizontal counter-current flow over the whole scaling region, different to the vertical heterogeneous counter-current flow.

A full-range drift-flux model was developed and verified against UPTF data at the GRS for the ATHLET code, Austregesilo et al., 2013. This model also is the basis for the interfacial shear model in ATHLET for both the horizontal- and inclined-pipes. This shows how the scalability of the ATHLET system code has been solved with respect to the flow conditions in the hot leg during reflux condenser mode.

The UPTF test demonstrated that a substantial margin exists between the flooding limit and the typical conditions expected in a PWR during reflux condenser mode of a small-break LOCA, Fig. 4-17. This mass flow of steam results from a 2% core-decay power, scaled from 8 MPa to the UPTF’s test pressure (0.3 MPa).

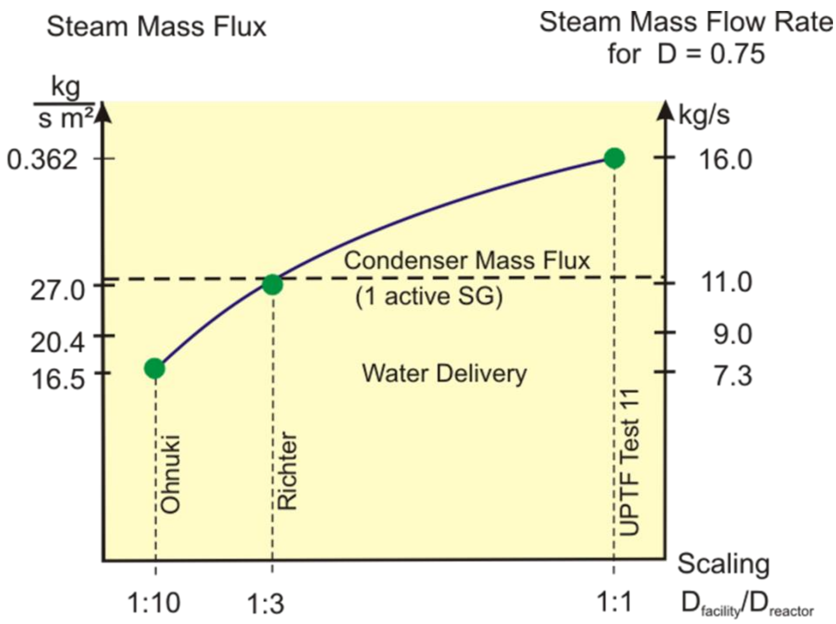


Fig. 4-17 – Counter-current flow limitation correlation compared with reactor steam's mass flow in the hot leg.

4.3.5 Issues related to applying the code to an NPP

Scaling issues related to application of the code to a NPP may exist if the scalability of the code is not sufficiently proven with respect to dominant process. This may occur in the main following cases:

- The code has not demonstrated good quality in calculating IET tests relative to the transient of interest at different scales when comparative tests exist.
- Some phenomena were distorted in IETs tests relative to the transient of interest, and the code is not proven to have good predictive capabilities for these phenomena at the NPP scale by validation against appropriately scaled SET data.
- A sensitive local phenomenon is modeled by a closure law that has poor scalability and induces significant errors at the NPP's scale.
- Some inherent limitations of the code models associated with non-modeled physical processes appear more sensitive at reactor scale than in all validation cases, and induce significant errors.
- Some nodalization issue induces errors at the NPP scale.

Most of these possible issues may be avoided or reduced to an acceptable level by the following:

- An exhaustive and properly applied PIRT method that identifies all dominant phenomena at system-, component-, and process-levels,
- an extensive validation of the code against the appropriate SET- and IET-data covering all dominant processes: if some significant differences in the predictions compared to experimental findings are found, they must be taken into account in the uncertainty quantification,
- A sufficient analysis of any nodalization issue,
- checking that any issue of code scalability is accounted for in the UQ for the reactor's application.

4.4 Scaling and uncertainty quantification in BEPU

Nuclear Reactor Safety (NRS), NPP safety review, NPP accident analysis, Safety Assessment including Deterministic (primarily) and Probabilistic branches (DSA and PSA, respectively), Best Estimate Plus Uncertainty (BEPU) approach, Country-specific NRS Regulatory documents, and recommendations and reports issued by International Institutions like the IAEA and NEA, all are of concern here to identify the role of, and the objective for scaling. Even the introduction of comprehensive definitions for each of the contexts associated with the listed topics is outside the purposes for the S-SOAR. However, the connection between scaling and the NRS framework is outlined in section 4.4.1. A synthesis of current (available) experience in connecting scaling and safety requirements is provided in section 4.4.2, and Section 4.5 is a proposal for the consistent exploitation of scaling concepts and findings within the BEPU approach.

4.4.1 The BEPU approach

NRS shall be seen as the general framework for applying scaling-related concepts. It may be seen as consisting of two key parts: A) The requirements on the one side, starting from the safety objective to minimize the radiological impact; and, B) The Nuclear Installation (noticeably the NPP) as built and operated, i.e. the Systems Structures and Components (SSC) and the software, e.g. control logics, designed and installed to comply with the requirements. The established notion of Defense in Depth (DiD) is in-between the NRS parts A) and B). The DiD concepts become concrete with the characterization of Prevention and Mitigation, and through the identification and the classification of Safety Functions, Safety Barriers, and Engineered Safety Features (ESF). Principles, concepts, or actions such as ‘fail-to-safe’, ‘As Low as Reasonably Achievable’ (ALARA), ‘redundancy’ and ‘minimizing or avoiding Common Cause Failure’ and frameworks like the Design Basis Accident (DBA) Envelope, also are functional in connecting the parts A) and B) of the NRS.

The Safety Analysis, or better, the Safety Assessment, is used to demonstrate the compliance of the NPP’s design and operation [part B) or NRS] with the safety requirements [part A) of NRS]. Furthermore, the probabilistic safety-assessment, PSA, uses deterministic calculations – i.e. the place where BEPU and scaling appear – to evaluate the consequences of envisaged accident conditions. Otherwise, the deterministic safety analysis, DSA, constitutes the key context for applying BEPU and scaling. DSA directly is used in evaluating the safety system (see below) in the framework of the DBA envelope, including the related acceptance-criteria.

Licensing shall be seen as the legal-, or regulatory-authority controlled part of NRS, and the FSAR is the compendium for safety assessment, e.g. for demonstrating the acceptability of any nuclear installation within the Licensing framework.

The FSAR includes the results of DSA calculations that shall be carried out following acceptable specifications and procedures. The pivotal chapter for the FSAR is Chapter 15 that deals with accident analysis. Several connections and mutual feedbacks can be identified between Chapter 15 and other chapters of the FSAR. Without entering into detailed descriptions, one may note that connections exist among the following topics: The safety aspects for core design (Chapter 4 of FSAR), the qualification of computational tools based on the specific NPP start-up test results (Chapter 14), the design or the demonstration of acceptability of technical specifications for systems, components, or logics of the NPP (Chapter 16), and the PSA results (Chapter 19). Hereafter, the relationship between the BEPU and scaling is outlined, consistently with the requirements for issuing FSAR, Chapter 15.

With reference to Accident Analysis (FSAR Chapter 15) the Licensing history, the current status, and the perspectives can be summarized by two sets of documents, D’Auria, 2012, the former dealing with the requirements issued by the USNRC, the latter constituted by supporting technological information issued by the IAEA and the NEA (the year of issuing and main topic are reported where applicable; the arrow ‘➔’ implies innovation):

– Interim Acceptance Criteria for ECCS (1971) integrated into 10 CFR 50.46, and supported by the related Appendix K, e.g. see USNRC, 2014, (updated document),➔ RG 1.157, USNRC 1989a, ➔ RG 1.203, USNRC, 2005.
– NEA bases for validating the system's thermal-hydraulic code, SOAR on TECC (1989), SETF-CCVM and ITF-CCVM (up to 1996), i.e. refs. NEA/CSNI, 1989, NEA/CSNI, 1993, and NEA/CSNI, 1996, respectively, ➔ development and qualification of uncertainty methods, USNRC-CSAU NEA UMS (1997) and NEA BEMUSE (2010), i.e. NEA/CSNI, 1998a, and NEA/CSNI, 2011, respectively, ➔ methods, approaches, and framework for Accident Analysis, IAEA SRS 23, IAEA SRS 52, and IAEA SSG-2 (period 2002-2010), i.e. IAEA, 2002, IAEA, 2008, and IAEA, 2010, respectively.

As already mentioned, even a short summary of those documents is beyond the purpose of the S-SOAR; rather, we note that scaling constitutes a key issue in all of the listed documents. Furthermore, the following rings-of-a-chain, Fig. 4-18, always making reference to Accident Analysis, may be identified where scaling has a role.

BEPU constitutes the end product of about four decade's worldwide R & D projects. BEPU also requires a variety of codes (such as areas of neutron physics, nuclear fuel, containment, CFD type, structural mechanics, see also section 4.5) to be pursued, e.g. to complete the Chapter 15 of the FSAR. The key elements for BEPU are the best-estimate computational tools and the uncertainty methods, e.g. D'Auria et al., 2012. Sample, snapshots-items showing the importance of scaling in BEPU are constituted by the demonstrations as the following ones:

✓ A code is qualified against scaling, i.e. that code's capabilities are not affected (or are affected in an acceptable way) by the scale of the ITF where the validation data were measured.
✓ An input deck (nodalization) is qualified against scaling.
✓ The uncertainty method can bound the code errors expected in applications of the code at full scale (i.e. at the NPP scale).
✓ Procedures and databases exist that are suitable for the demonstrations above (more details are given in sections 4.1, 4.2 and 4.4.2).

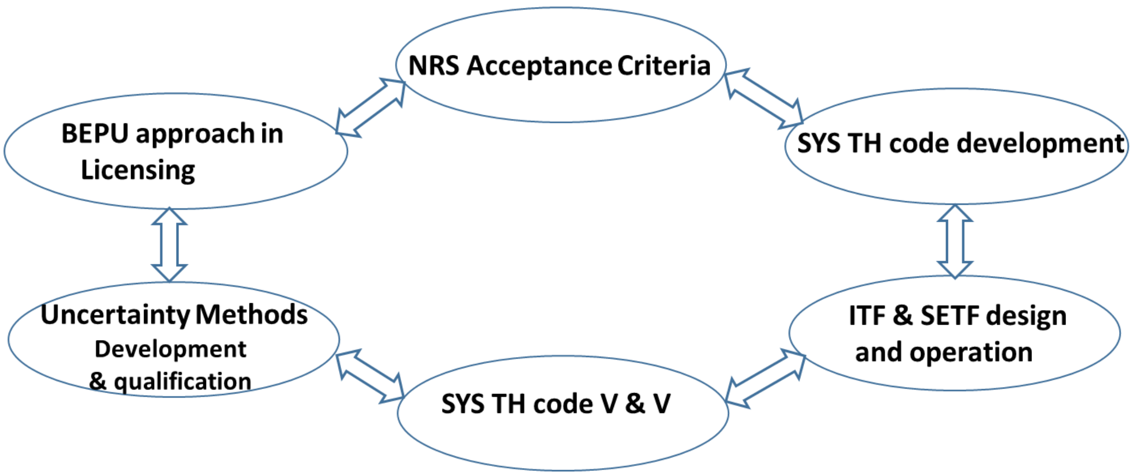


Fig. 4-18 – An overview of the Accident Analysis process in NRS.

4.4.2 The scaling in available UQ methods

The prediction of the uncertainty which characterizes the results of the calculations results by system thermal-hydraulic codes is one of the two key elements of BEPU; the other one is the code itself. There are detailed reports issued by the International Institutions dealing with uncertainty methods and qualification, such as NEA/CSNI 1998a, NEA/CSNI, 2011, and IAEA, 2008. It is not the object herein to duplicate the information in those documents, or even to summarize the key findings. Rather, the objective is to outline the connection between scaling and uncertainty so to demonstrate that scaling is definitely considered as a source of uncertainty in predicting the conditions in the NPP transient by the system's thermal-hydraulics codes.

To achieve this objective, three established procedures are discussed:

- CSAU is the pioneering procedure, including the uncertainty roadmap widely adopted by industry;
- the UMAE-CIAU methodology as the prototype uncertainty-method based upon the propagation of output errors in the calculation;
- the GRS methodology as the prototype uncertainty-method based upon the propagation of input errors in the parameters.

4.4.2.1 Scaling in CSAU

CSAU is a 14-steps procedure, e.g. USNRC, 1989, starting from identifying a scenario or transient, power plant, important phenomena based on Phenomena Identification and Ranking Table (PIRT), identifying code models, and establishing the code's applicability. In the CSAU procedure, three uncertainty sources are identified that are to be quantified – the code and experiment accuracy (Step 9), the effect of scaling (Step 10), and reactor's input parameters and state (Step 11). CSAU ends with the estimation of total uncertainty. Figure 4-19 is a flowchart of the step 10 of the methodology.

The Step 9 is about estimating the code's uncertainty based on experiments, and Step 10 is about determining the effect of scaling, and the resulting bias and uncertainty. Normally, these two steps have been combined. With them, the uncertainties of code models are estimated and are input to Step 13 where all the uncertainties and biases are combined. CSAU Step 9 is about comparing the code's prediction of the figure of merit with the measured value to estimate uncertainty. There are two approaches of estimating uncertainties, based on using integral effects tests, or separate effect tests. If there are well-scaled integral facilities for the transient of interest, a comparison of the predicted- and measured-values of the figure of merit can provide an estimate of uncertainty. However, it is very difficult to design a single integral-effects facility that can scale the transient for all phases. Also, as the uncertainty propagates with time, the initial lower-scale uncertainty will grow over time. Therefore, in general, the role of integral facilities is to verify an estimate of the overall uncertainty in the figure of merit attained by propagating uncertainties through the code and combining them, as is done in the second approach. In this approach, the code uncertainty is estimated through the combination of uncertainties and biases of different phenomena as computed from separate-effects tests. This overall uncertainty in figure of merit should always be higher than the uncertainty estimated from integral-effects test facilities. In the remaining documentation, the second approach is described, as applied to LBLOCA.

In the CSAU approach, the uncertainty in predicting phenomenon is estimated by identifying the appropriate parameters to represent it, and selecting the separate-effects tests. In cases where there are data for full-scale facilities with reactor thermal-hydraulic conditions, modelling the tests with the proper nodalization, comparing the predictions with the data, and accounting for instrument uncertainty, will suffice. However, for small tests, a scaling study will be needed to find scaling bias, or counterpart tests must be used at different scales. Engineering judgment and sensitivity studies also are used to estimate the uncertainty range in the designated parameters representing the phenomenon, or the bias in the figure of merit.

The scaling study is performed in Step 10, which is the focus of this project (Fig. 4-19 shows the details). The information obtained from Steps 1 to 5 in the CSAU methodology, i.e. the PIRT results and code-assessment manual offer the starting point for Step 10 of the evaluation. The uncertainties and biases to be obtained therein fall into two categories: 1. Evaluation of test facility's scaling distortions on important processes; and, 2. Evaluation of the scale-up capabilities of the closure correlations used in the code. If the distortion of the test facility impacts important processes (in the first category), or the scale-up capability of the correlation in code affects the simulation (in the second category), appropriate uncertainties and biases must be specified. The detailed sub-steps in Step 10 can be referred to in Appendix C, and examples are given in Appendices L, M, N, and O of the CSAU methodology. A bias was estimated for the scaling effect of ECC bypass phenomenon, Rohatgi et al., 1990.

In estimating the uncertainty or bias induced by scale distortion in an integral test facility, the impacted parameter (e.g. peak cladding temperature in a LOCA) and the dominant contributing parameter (e.g. linear rate of heat generation in LOCA) are plotted for all available data points covering all scales. Then, the uncertainty in reaching the 95% confidence level is used as confirmation of the CSAU procedure of aggregating important uncertainties in phenomena, resulting in total uncertainty in the FOM.

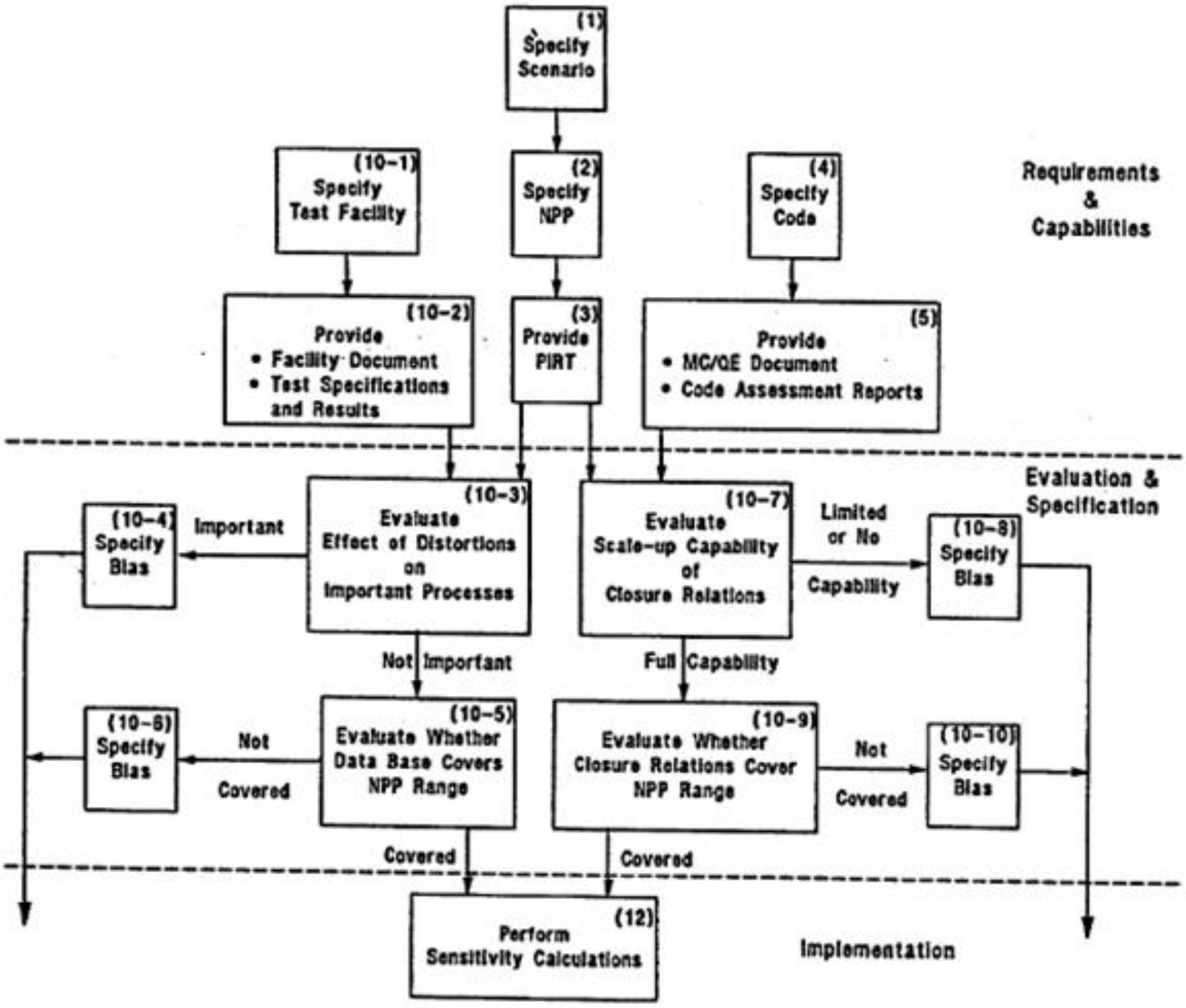


Fig. 4-19 - Step 10 of CSAU Methodology

In evaluating the scale-up capabilities of correlations in the code, based on the PIRT result, we evaluate the processes that impact the parameter of interest e.g. the model of the reactor's coolant pump, the critical two-phase flow through breaks, and liquid entrainment to steam generators in a LOCA. All available scaled data used to develop the correlation or the models in the code are compiled to determine the uncertainty or bias to reach a 95% confidence level.

If the distortion and scale-up capability are satisfied, the range of the NPP data needs to be compared to the conditions of the database experiments, and the closure correlations in the code. If the NPP range is not covered, additional biases are needed. After this evaluation, all the uncertainties and biases are added together as the bias from scaling (Step 10). Both are combined with other sources of uncertainty from Steps 9 and 11 into the total uncertainty or bias (Step 13) for sensitivity calculation (Step 14,) using the NPP nodalization.

4.4.2.2 Scaling in UMAE-CIAU

The UMAE logical diagram is given in Fig. 4-20. One significant connection between UMAE and scaling already was discussed in Section 2.1.3.4 (scaling achievement). Therefore, the information in the following should be integrated with what is provided in Section 2.1.3.4, related to scaling. More details are found in the UMAE reference literature, e.g. D'Auria et al., 1995, and IAEA, 2008. Hereafter, the bases of UMAE are recalled to emphasize the connection with scaling.

Historically, UMAE originated from the scaling study of the PIPER-ONE facility, a BWR, experimental simulator for SBLOCA, Mazzini et al., 1981, and the capabilities (including the deficiencies) of system thermal-hydraulic codes. The idea (of UMAE) came from two dead-ends of research studies in the area of accident analysis for LWR:

1) Experimental data measured in ITF cannot be extrapolated to the NPP because of unavoidable distortions in the design and construction of the ITF. In addition to the examples provided as scaling achievements in section 2.1.3, there are heat releases from structures to the coolant, and heat losses to the environment: it is impossible to avoid either one or both of the (typically severe) distortions, no matter what scaling laws are adopted.
2) Calculated data by the system's thermal-hydraulic code, i.e. using input decks (or nodalizations) of different sizes, cannot be extrapolated to the NPP. In fact, there is no guarantee that physical models and numerical methods can maintain accuracy in the results when the size, and typically flow area, of the nodes is changed.

This led to the idea of the UMAE (or the error scaling procedure): the error scaling procedure is based on connecting experimental data and the corresponding calculated results. In this procedure, the database is constituted by time trends of relevant thermal-hydraulic variables measured in the ITF with different scales and by results of 'qualified' (not 'tuned') code-calculations. If all of the following conditions are met, then the error which demonstrates a random characteristic affected by many uncertainty or error-sources can be extrapolated to the NPP conditions, see also D'Auria et al., 1988:

a. The measured trends are suitable to characterize the phenomena, including properly qualifying the originating signals,
b. A suitable number of experiments (counterpart tests) are available at different scales (range 3 to 10) where the same phenomenon is detected;
c. The range of scales of ITF is sufficiently broad, e.g. such that the range covered by the experiments is larger than, or comparable to the scale-range which separates the largest ITF from the NPP;
d. The error in code prediction (assuming there is a suitable procedure for developing qualified input decks without tuning) is not affected by the scale or by the scaling laws adopted for the design of the ITF;

e. There is no bias in, nor tendency of the code to over-predict or under-predict an assigned variable (in that case, the code needs additional development before being applied with UMAE);
f. 'All' the phenomena measured in each ITF are reproduced in the code calculations with 'acceptable' errors; the acceptability of errors shall be evaluated at the qualitative- and quantitative-levels, Bonuccelli et al., 1993; at a quantitative level the FFTBM is used: for the development of the FFTBM, see Ambrosini et al., 1990, see also Kunz et al., 2002, for the application see D'Auria et al., 1989, D'Auria et al., 1994, Prosek et al., 2002, and Prosek et al., 2006;
g. Data from large-scale SETFs, and NPPs, even a limited number of phenomena or narrow variation of the parameters, do not contradict the findings, i.e. in relation to the code's prediction capabilities, related to smaller facilities,

In the case of UMAE, the lack of experiments prevents the possibility of scaling of the error: so, no experimental data = no error in code calculation = no possible application of system thermal-hydraulic codes within NRS.

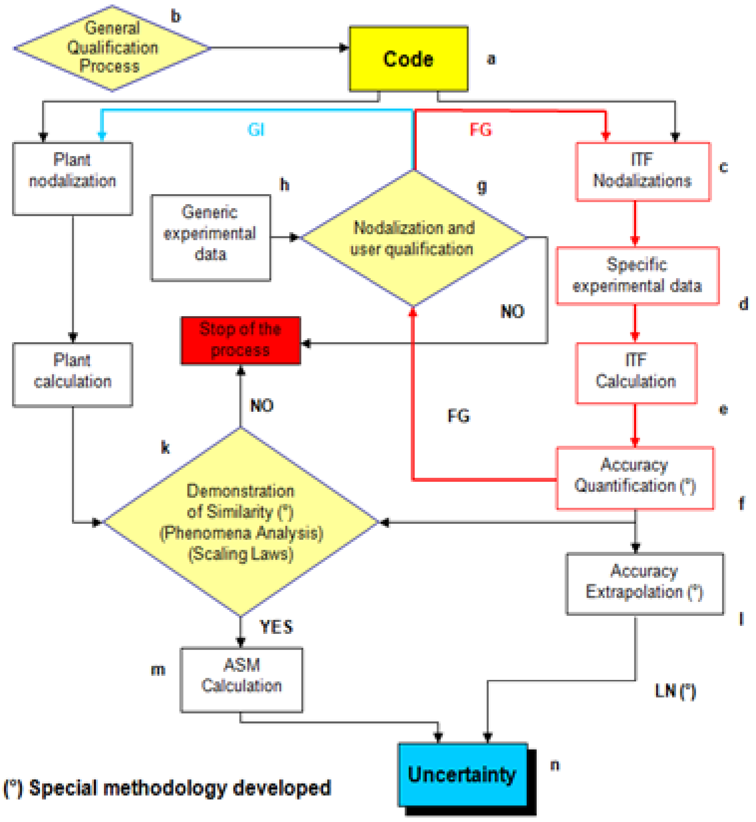


Fig. 4-20 – Logical diagram of UMAE

The red loop on the top right of Fig. 4-20 is a working path wherein the conditions listed above are met; further details can be found in the paper by D'Auria et al., 1995. Namely, the possibility of demonstrating the capabilities of the code-user, as well as the code's robustness, and the demonstration of suitable level of quality of the input decks also are part of those conditions.

After a successful exit from the red loop, the code user may start developing the input deck for the prototype NPP, based on the experience learned from simulating the ITF (proper conditions also apply to this process).

Then, the other key-scaling-related step of UMAE is the demonstration of similarity between NPP prediction (left side of the diagram), and one set of ITF experimental data (right side of the diagram). This is achieved throughout the so called ‘Kv-scaled’ calculation (see Section 4.2.4): typically, an NPP calculated accident-scenario is expected to differ from the ITF measured scenario owing to unavoidable scaling distortions. However, all discrepancies between the NPP calculated results and the ITF measured time trends must be explicable in terms of scaling, and by performing proper sensitivity calculations. Completion of this step allows the full application of the UMAE for predicting uncertainty.

All the steps in UMAE are performed in the CIAU, D’Auria & Giannotti, 2000. This also includes the distinction between errors in time and quantity. In the case of CIAU, the connection between the uncertainty evaluation and the scaling remains as in the UMAE; furthermore, the concepts of a ‘quantity error matrix’ associated with ‘hyper-cubes’ defining the NPP’s status, together with the ‘time error vector’ are introduced. This directly connects the combination of variables ranges and the modelling error: e.g. a correlation for the heat-transfer coefficient is assumed to produce the same error in the range A-A1 MPa, B-B1 K, C-C1 void fraction, D-D1 heat flux, etc., whatsoever is the accident scenario which enters in those conditions.

In the case of CIAU, an automated procedure is set-up with the data set of calculated and measured trends available. This allows the filling of the quantity matrix (based on hyper-cubes) and of the time vector (based on time intervals) with accuracy/errors. The reverse process is regarding the use of errors. Each time the NPP calculation with a condition entering one hypercube or a time-interval, both quantity and time errors are generated which constitutes the uncertainty of a calculation.

4.4.2.3 Scaling in statistical uncertainty propagation approaches

The input uncertainties propagation methods for uncertainty analyses consider the effect of the uncertainties of input parameters, like computer-code models, the initial- and boundary-conditions and other application-specific input data and solution algorithms on the calculation results. In this way, it is similar to CSAU. The most widely used method of this type is the methodology proposed first by GRS, and based on well-established concepts and tools from probability calculus and statistics. The main advantage to using these tools is that the number of calculations is independent of the number of uncertain parameters to be considered. The necessary number of code calculations is given by the Wilks’ formula, Wilks, 1941. The number of calculations depends only on the chosen tolerance limits or intervals of the uncertainty statements of the results. Wilks’ formula does not need any assumption about the distribution of the result. Not using these statistical tools necessitates a drastic increase in code calculations with the number of uncertain parameters when the uncertain values of the selected parameters have to be combined.

The simplest approach consists in choosing combinations of parameter values at will, and performing the respective code runs, instead of applying statistical tools; this approach does not allow making quantitative tolerance / confidence statements about the combined influence of the identified uncertainties. Such statements could not be made without the tools from statistics, not even after a very large number of code runs. It simply is a matter of efficiency to exploit what is known from statistics to reach the coverage of target uncertainty and confidence at minimum cost, or to extract the most of information from the expended number of runs.

Description of the method based on Wilks’ formula (GRS type method)

In the statistical methods of uncertainties evaluation based on Wilks’ formula, the following steps are performed.

Identification of uncertainties

First, the method requires identification of the potentially important contributors to the uncertainty in the code results. These contributors consist of

- uncertain model parameters,
- uncertain initial- and boundary- conditions (initial plant state),
- uncertain geometry (e.g. bypass-flow cross sections),
- uncertain scale effects.

Modelling uncertainties are represented by additional uncertain parameters. These represent two possibilities:

- adding on, or multiplying correlations by a corrective term,
- a set of alternative model formulations.

Then, the respective state of knowledge is quantified by subjective probability distributions. The term „subjective“ is used here to distinguish between subjective probability due to imprecise knowledge, and probability due to stochastic- or random-variability. This does not preclude stochastic variability from serving as a basis for quantifying the state of knowledge wherever appropriate. Such a distribution expresses how well the appropriate value of an uncertain parameter of the code application is known in the light of all available evidence. A state of knowledge based on minimum information at the parameter level is expressed by uniform distributions.

If parameters have common contributors to their uncertainty, the respective states of knowledge are dependent. This dependence needs to be quantified, if judged to be potentially important. Measures of association (correlation coefficients), conditional subjective probability-distributions, and other means are available for this quantification.

Uncertainties in geometry discretization, to describe the important phenomena, are to be taken into account and should be optimized in the code-validation process. However, if geometry discretization uncertainties are included, alternative nodalization schemes can be included in the uncertainty analysis. In general, the results of code validation are a fundamental basis for quantifying uncertainties in input parameters.

This analysis of input uncertainties clearly requires expert judgement to varying degrees, while the steps of the actual analysis largely are mechanistic and well founded in probability calculus and statistics. The selection and quantification of these uncertain parameters are based on experience gained from validating the computer code by comparisons between the model’s predictions and test data of integral tests and separate effects tests for the model parameters, as well as on known uncertainties.

Selection of important uncertainties

All potentially important uncertainties (model parameters, initial- and boundary-conditions, reactor plant’s operating parameters, material properties, nodalization, numerical parameters and scaling effects) are selected. There is no limitation to a small number of uncertain parameters since the number of code calculations does not increase with their numbers.

Quantification of uncertainties

For each of the selected individual uncertain parameters, subjective probability distributions are specified to quantitatively express the corresponding state of knowledge. This means, in addition to the uncertainty range, that the knowledge is expressed by subjective probability-density functions or probability distributions. This is to account for the fact that evidence from previous code validation or experimental evidence indicates that the appropriate value of a parameter is more likely to be found in certain sub-ranges of the given range than in others. The probability distribution is called “subjective“, since it expresses the state of knowledge of fixed but unknown or inaccurately known parameter values, rather than their stochastic variability. The classical interpretation of probability as the limit of a relative frequency, expressing the uncertainty due to stochastic variability, is not applicable here. Important for quantifying uncertainties are the results obtained during the process of validating the computer code. An important source will be the separate-effects test and the integral test facility's validation matrices.

Qualification

Qualification should be considered separately for the computer code, input deck (including nodalization), code user, and user of the uncertainty method, as well as the data base used for quantifying model uncertainties. In particular, the input deck should be carefully qualified. Usually, in the uncertainty analyses the input deck is recognized as qualified and no uncertainties related to it are considered.

Combination and propagation of uncertainties

Input uncertainties are represented by uncertain parameters and quantified by the probability distributions expressing the state of knowledge. A random sample is drawn according to the specified parameter distributions, as well as to any quantified state of knowledge dependences. All uncertain quantities are varied simultaneously for each code run. An element of this sample is called a parameter vector, and it comprises one value for each of the uncertain quantities. The code is run with each parameter vector in the sample, and the set of output values obtained again constitutes a random sample, but one that was drawn according to the unknown subjective probability distribution of the respective code results. From this sample, quantitative uncertainty statements are immediately derived by applying statistical concepts and methods.

The minimum number of code calculations depends on the requested probability content, and confidence level of the statistical tolerance limits used in the uncertainty statements of the results. The required minimum number of these calculation runs is given by the Wilks' (1941) formula. The uncertainties propagation method as applied by GRS, Hofer, 1990, relies on actual code's results without fitted response surfaces or other approximations like goodness-of-fit tests.

Scale-up effects in methods based on propagating input uncertainties

The correct way of considering the scale-up effect in uncertainty analyses appears to be performing independent uncertainty analyses for different scales. The scale-up effects are to be considered by developing a qualified input data set for each reference calculation (best estimate) and by selecting and quantifying uncertain input parameters. In particular, differences in uncertainties in the physical models according to their application to different scale objects have to be accounted for. In the uncertainty analyses performed by GRS differences in model uncertainties by their application to small-scale test facilities and to large- scale test facilities or NPPs usually are considered. These differences mainly are expressed by different uncertainty ranges. Mostly, such differences have been identified for the closure relations of the conservation equations. For instance, for the geometry of the vertical annulus (geometry of the downcomer) in large-scale facilities, a much wider variation range of interfacial friction for ATHLET applications was established than in small-scale facilities. Analyses of the LB LOCA accident at the Zion NPP showed that the variation range of interfacial friction in annulus geometry was 0.05 – 3.0, Skorek, 2009. A similar range was applied for analyses at the middle-scale annulus in the LOFT test facility. For a small-scale annulus, much narrower uncertainty ranges are applied. For instance, for LOCA analyses at the LSTF test facility, the applied uncertainty range was 0.33 – 3.0, Skorek et al., 2011. The dimensions of the annulus in the LSTF facility are much smaller than the dimensions of the annulus in real reactor, such as in the Zion or even in the LOFT facility. The aim of the extension of the variation range is to reduce the interfacial friction in the ATHLET. It takes into account that the 1-D interfacial friction correlation for annulus geometry in the ATHLET code was developed on the basis of small-scale experiments. The relatively low value for the large-scale geometry results from different behaviour of flow in the large-scale and in the small-scale facilities. In particular, CCFL is much less restrictive for large scale annulus than for small scale annulus. Also for other interfacial friction correlations, like interfacial friction in bundle geometry or in pipe geometry, different ranges of variation are applied for small- and large-scales.

Other possibility to consider in scaling effects is the selection of different correlations according to their field of application. If there is such option in the physical model of the thermal-hydraulic code, different correlations/constitutive equations may be applied for small- and large-scale facilities according to the recommendation in the code's documentation. Such recommendations result from the development and validation of the code, and as such express the state of knowledge concerning also the scaling effect.

The variation of the uncertainty range is the main way of considering scale-up effects in uncertainty analyses. Since the quantification of model uncertainties takes place by comparison with experimental data, an appropriate selection of the adequate experiments is of importance. Quantifying the preferable model uncertainties quantification lies in comparing code predictions with experimental data from separate-effect tests. The experimental data selected for quantification have to be representative of the considered application. In particular, they have to reflect the scale of the analysed facility.

Another possibility for quantifying model uncertainties is the estimation of general model uncertainty for the whole field of possible applications for small-scale facilities as well for large ones. However, in such a case, the estimated uncertainties in the models usually would be larger (or in the best case, the same) as for scale-dependent quantification. This would lead to a less meaningful determination of uncertainty limits.

The analyses of small-scale facilities could be used for identifying potentially important uncertain input parameters. However, for each application, a careful identification and selection process must be performed. As it has been found in such studies, Skorek, 2009, and NEA/CSNI, 2011, that very different parameters may be influential for small- and large-scale facilities.

In this context an approach considering all possible input uncertainties seems to be advantageous compared with the application of a Phenomenon Identification and Ranking Table (PIRT). The identification and limitation of parameters are carried out by experts on the basis of their experience. Since there are many smaller facilities than large scale ones, the knowledge of phenomena in small-scale facilities is much better than in the large ones. As a result, some important parameters in the large-scale facility might not be identified, simply because they never appeared as influential as in small-scale ones.

The condition for a correct consideration of scale-up effects is to carry out carefully complete uncertainty and sensitivity analyses for each application. The results of uncertainty analyses for small-scale facilities are important source of information and experience, but cannot be directly transformed to large-scale applications. The most important step to be performed in order to consider the scale-up effects is the identification and quantification of input uncertainties, in particular, model uncertainties for large-scale applications. Since some large-scale separate-effect experiments exist, the quantification can be performed in the best way on the basis of comparison with the experimental data. Once the model uncertainties have been quantified, the propagation of the input uncertainties through the mechanistic codes enables carrying out best estimate plus uncertainty analyses for any transient or accident in the field of the code's application. Also for event for which integral tests does not exist, this capability is a clear advantage of the uncertainty-estimation method based on Wilks' formula. However, it requires a proper quantification of input uncertainties and sufficient experimental basis of separate effect tests for model uncertainties quantification.

4.5 A Scaling road-map

Scaling plays important role in safety analyses, the experimental data base, the practice of BEPU approaches, system codes and related uncertainty methods. There is a need to address scaling issues in safety review process with all the available data, tools, methods, and approaches. A scaling roadmap is a good way to group these subjects together. Due to different approaches of BEPU and uncertainty methods, there are different ways to achieve the same goal (viz., to meet safety requirements). In this report the group includes two scaling roadmaps to demonstrate different approaches. The first one is based on the application of CSAU, and the other is proposed by D'Auria & Galassi, 2010. These two roadmaps can be seen as an overall methodology combining scaling analysis, experimental data, code calculations, and uncertainty methods to fulfill safety requirements of NPP as discussed in section 4.4. The common background for setting up the scaling road-map is based on the following assumptions:

– A system thermal-hydraulic code is applicable to perform safety analyses: namely, this is one way to demonstrate the compliance between the design-operation of NPPs and the established safety goals;
– The BEPU approach is pursued in applying a system's thermal-hydraulic code to perform safety analysis so to meet safety requirements;
– Use of best practices, databases, and information from reports issued by international institutions (namely NEA and IAEA), and of applicable computational tools and related application procedures to conduct safety analyses.

4.5.1 Scaling roadmap based on the application of CSAU

Scaling is an important process for designing test facilities at smaller size, power, and possibly pressure to accurately simulate the phenomena of interest expected to occur in a nuclear-power plant under abnormal conditions. These test facilities provide essential information for designing power-plants to achieve a specified performance and to assess the efficacy of safety systems for new plants, or existing ones. The actual determination of the performance of power plant and of safety systems is done through system codes designed for this purpose. The fidelity of predictions is estimated by aggregating the contributions of uncertainties from code models, nodalization, numerics, user options, and approximations of the power-plant's representation to fit code requirements.

Figure 4-21 shows a simple road map that captures the role of scaling in determining the fidelity of code predictions for nuclear-power plants for specific abnormal conditions or hypothetical accident scenarios. This road map is based on CSAU.

The road map shown in Fig. 4-21 has eight steps. The first step is to define the problem by specifying the NPP, transient, and the system code that will be used. The second step is to define the figure of merit that will ensure the safety of power plant by preventing any release of radioactive material. It could be the peak clad temperature, or the mixture level in the core. Both will ensure that the first barrier of radioactive materials remains intact.

The third step is to identify phenomena that have most impact on FOM. This step is to create a phenomena identification and ranking table (PIRT). This step is based on expert opinion, sensitivity studies, or scaling.

The fourth step is to assess the code's applicability for its intended application based on the important phenomena identified in PIRT from Step 3. In this step, the applicability of code formulation and constitutive relationships are reviewed. Many constitutive relationships are empirical and are from small-scale tests and are characterized by limited ranges of conditions for validation.

The fifth step is to identify tests, both separate effects, and also integral effect tests based on the PIRT. The scaling basis for these tests also is evaluated and limitations are identified. In addition, counterpart tests are also identified as they are the key for any extrapolation.

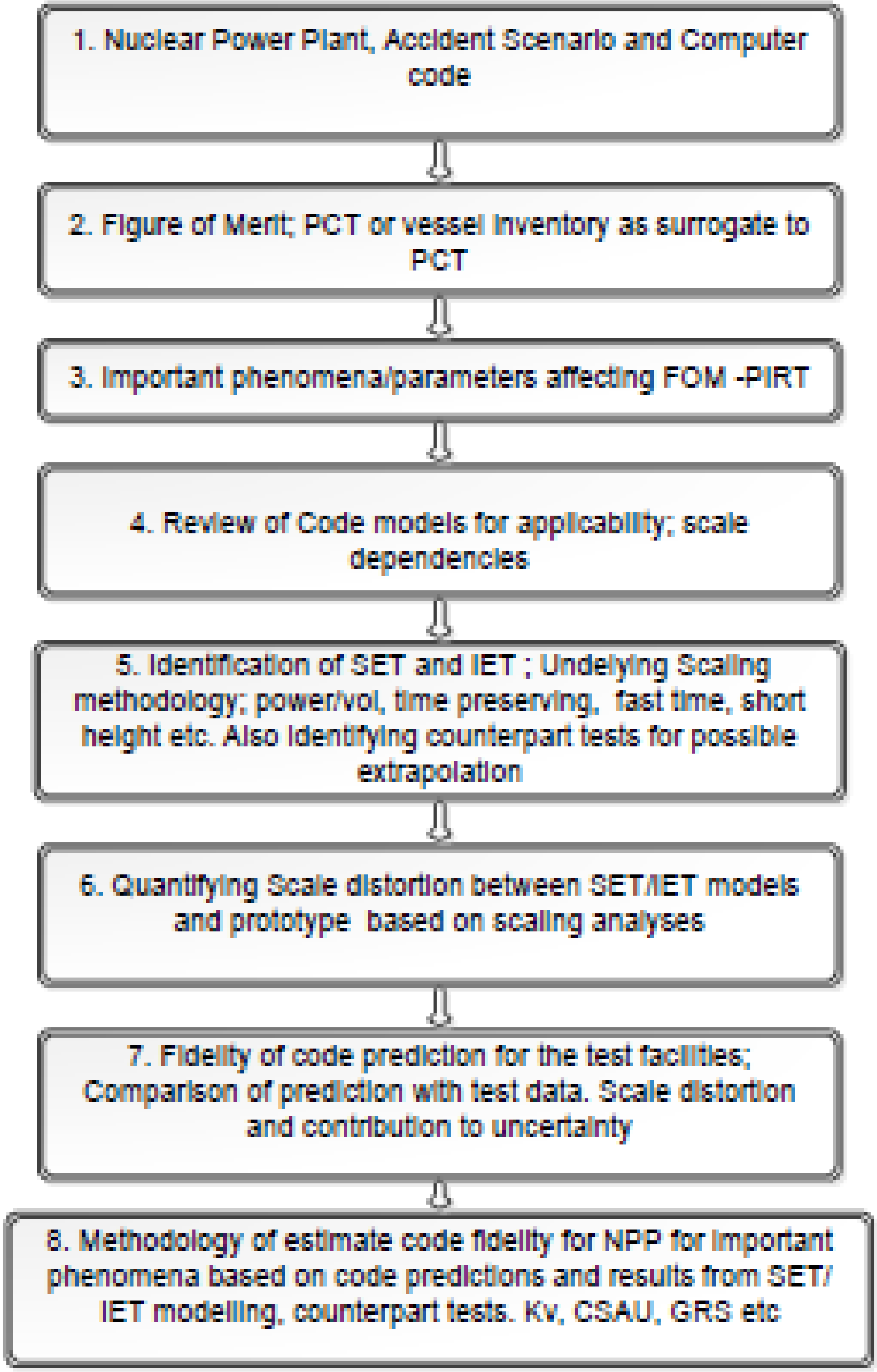


Fig. 4-21 – Scaling roadmap based on the application of CSAU.

The sixth step is to determine scale distortions. There are two ways to address this: the first is scaling analyses, and comparing non-dimensional groups from NPP and facilities, and the second approach is to model the test facility and NPP, and compare the parameters representing the phenomenon of interest.

The seventh step is to determine the fidelity of code predictions for SET and IET. The important group of tests is either the one that represents a phenomenon at almost-full scale, or the one represents one phenomenon at different scales. This information feeds in the eighth step where overall accuracy of the code is determined.

In the eighth and the last step, all sources of information from previous steps are combined to make one statement of uncertainty in predicting FOM. TH codes provide the integration and propagation of individual model uncertainties. Many approaches are listed here in the map, and they use information differently to estimate uncertainty.

4.5.2 Scaling roadmap proposed by D'Auria & Galassi, 2010

Considering the present status of the safety analyses, the existing experimental data-base, the current practice of BEPU approaches, the maturity of system codes, and of related uncertainty methods, a roadmap (summarized below) for scaling in a licensing process was proposed by D'Auria & Galassi, 2010. This may be seen as a possible overall methodology and it illustrates how scaling analysis, experimental data, code calculations, and uncertainty methods may contribute to the safety analysis of a reactor transient.

The scaling road-map shall be seen as the procedure for consistently applying scaling to the licensing process of NPP, as was discussed in section 4.4. The background for setting up the scaling road-map is constituted by the following statement-assumptions:

- a system thermal-hydraulic code is an acceptable tool to undertake accident analysis: namely, this is the best viable way to demonstrate compliance between the design-operation of the NPP and the established safety goals;
- the BEPU approach is pursued in applying a system thermal-hydraulic code to the accident analysis (typically recognized as Chapter 15 of the FSAR);
- the NRS ALARA principle is translated into the area of safety assessment as "…the best use of practices, databases, information part of reports issued by international institutions (namely NEA and IAEA) and of (properly) qualified computational tools and related application procedures to perform accident analysis".

Having this in mind, the diagram in Fig. 4-22 was proposed, D'Auria & Galassi, 2010.

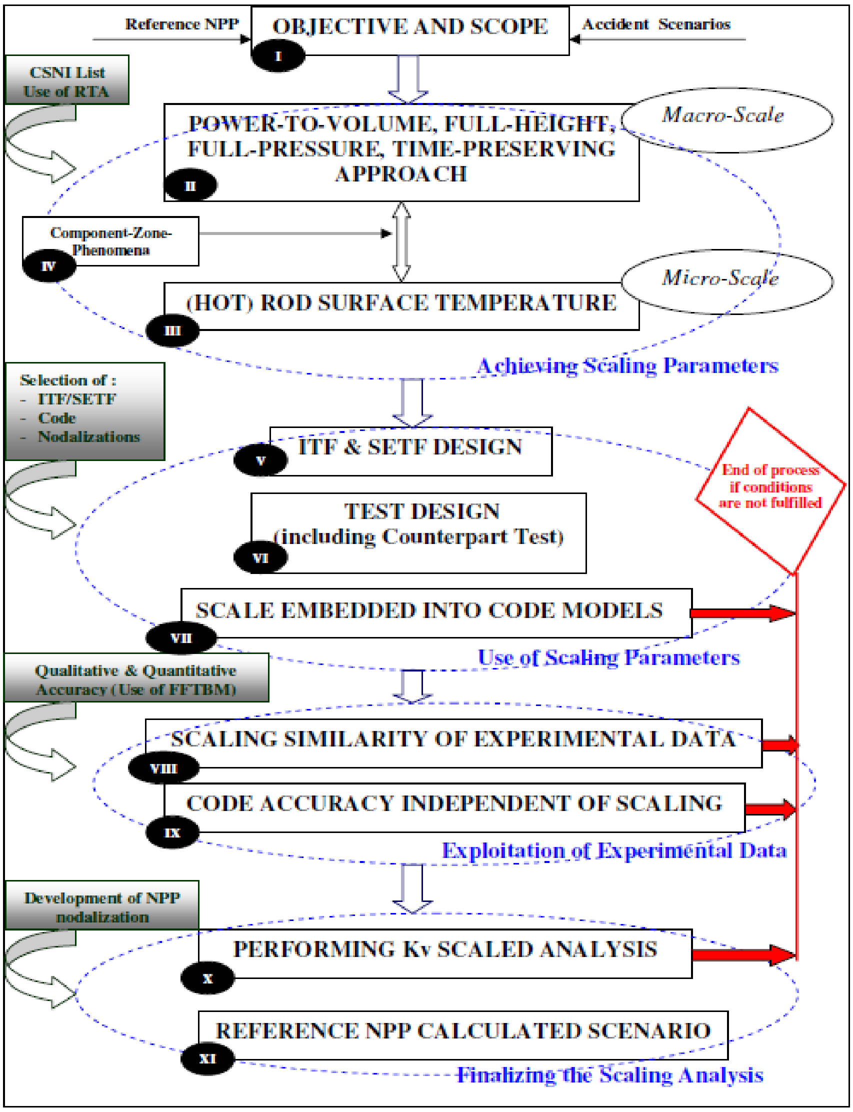


Fig. 4-22 - Roadmap for scaling in a BEPU-based NRS Licensing process.

As already mentioned, the BEPU approach implies the use of several computer codes and of related coupling techniques and software. Hereafter, attention is restricted to the application of system thermal-hydraulic codes.

As a first notation, it shall be noted that (almost) all the elements which are mentioned for the Scaling Database and Knowledge Management in Chapter 1 are part of the diagram of Fig. 4-22. This confirms that scaling is an established technology, and that the scaling road-map collects, makes use of, and integrates (all) the available scaling elements. Furthermore, qualitative- and quantitative-acceptability thresholds are part of the procedure (the red arrows in Fig. 4-22). Within a NPP safety review process, the acceptability thresholds are fixed by the licensor, or are proposed by the licensee and accepted by the licensor. No compliances with acceptance criteria imply the stopping of the procedure and the request for new data (either calculated and/or experimental) or, even additional R & D. The established road-map is expected to satisfy the requirements of a BEPU-based NPP safety-review process, even though this achievement can be established only by the Regulatory Authority. The roadmap is expected to be followed each time a new NPP safety review process is started; some steps are (obviously) common to different processes.

There are four macro-steps in the roadmap after the objective and the scope for the scaling activity are established. The objective and the scope include the selection of the NPP, and of the plant scenarios to meet the safety requirements when pursuing the BEPU approach. Furthermore, on the left side of the diagram (shadow blocks) support information for each macro-step is specified. The four macro-steps, blue ellipses in Fig. 4-22 are the following ones:

A. Achieve scaling parameters: The objective here is to derive by the best available scaling techniques, a set of parameters suitable for designing experimental facilities. The activities are to be performed in the beginning of the procedure. However, the resultant scaling parameters should be confirmed by a subset of the worldwide available ITFs and SETFs that complies with the objectives.

The reference (sample) support information includes the list of phenomena, e.g. OECD/CSNI/NEA, 1989, as well as PIRT results specific to the selected NPP and accident scenario, where only the phenomena identification part of the PIRT is needed.

B. Use of scaling parameters: two recognizable product types are to be generated in this step, i.e. the SYS TH codes and the ITF/SETF design with the related matrices of experiments. It will be demonstrated that both product types are consistent with the scaling parameters defined in the previous step.

The information on the reference sample here includes the CCVM for the ITF and the SETF, NEA/CSNI, 1993, and NEA/CSNI, 1996, and is supported by comprehensive code user manuals.

Acceptability criteria for the code models are established by the regulator, or proposed by the applicants; methods for this are mentioned below, (see also sections 4.1 and 4.2). In the case of facilities and tests, it is expected that available ITF and SETF and related experiments, including ranges of parameters, are acceptable to address the scaling issue. However, a specific report may be needed to confirm consistency between the scaling parameters decided upon in previous macro-step, and features of chosen SETF, ITF and of test-matrices.

C. Exploitation of experimental data: The transient-signals measured during experiments and the results from code calculations are evaluated here (the data on test design data and code capabilities are evaluated in the previous macro-step). Two key activities are conducted to demonstrate the following :

   – Phenomena and related representative time-trends gathered from experiments do not depend upon the selected set of scaling laws (or the scaling theories mentioned above) for the design of the ITF or the SETF. Scaling distortions, if identified, are characterized and justified. Only experimental data are involved herein.
   – Code capabilities are not affected by scale: Experimental and calculated data are of concern here. This demonstration is the same as that requested for applying the UMAE procedure (see section 4.4.2).

D. Finalizing the scaling analysis: The general objective here is to use data and information from the previous steps for the analysis of the plant scenario for the target NPP. This is accomplished in two steps:

   – Perform the Kv-scaled calculation, a required step in the UMAE procedure (section 4.4.2, see also section 4.2.4).
   – Verify that the phenomena predicted for the NPP transient at the beginning of the procedure are part of the experimental data collected. The agreement should include the ranges of the parameters of key variables and their combinations.

The procedure repeats for each test (at least each type of test) specified in the safety requirements. In the following, the individual steps labeled from I to XI in Fig. 4-22 are discussed in more detail. An attempt is made to avoid repetition of concepts already described in the macro-steps from A. to D.

I) Objective and scope: The adopted code, the NPP, and the target transient-scenario are identified and characterized here.

II) Scaling approach: Any approach described in Chapter 3 is applicable here. Among the best ones available, the full-height, full pressure, full linear power, and time- preserving scaling laws (using water as working fluid) are preferred for designing the ITF and SETF. This scaling approach plays an important role in the entire procedure.

III) (Hot) Rod Surface Temperature (RST): A variety of findings and requirements constitute the output of a scaling analysis. Several target variables and parameters are part of scaling parameters determined in the scaling analysis. However, ITF, and SETF that involve the transient simulation of RST with dry-out and rewet with a scaling factor for q” (wall heat flux in W/m^2) equal to unity are needed, and should be available in the data base in the procedure.

IV) The scaling of special components (e.g. pump, separator), zones (e.g. RPV downcomer, SG inlet chamber) and phenomena (e.g. two-phase critical flow, reflood, and natural circulation) should be performed here. A cross-connection table is expected to result from this step.

V) The design of test facility, either ITF or SETF, is the main activity. A key action consists in defining the size (i.e. volume and/or power, and/or total height, and/or pressure) of the facility and other significant parameters. The budget may become a main constraint in the design or the construction: This should be acknowledged by the designers (who should not choose scaling laws to compensate for lack of funding). Any geometric parameter (e.g. equivalent diameter, height and elevation of any individual component) or any operating parameter (e.g. pressure drop distribution, nominal pressure, mass flow-rate) should be characterized and compared with the (ideal) value of the scaling factor resulting from the activity at macro-step A., above. A design scaling report should be issued where the distortions are characterized, as well as the expected consequences upon the transient scenarios.

VI) The design of the ITF or the SETF typically accompanies a (minimum) set of experiments, i.e. the test matrix. Experiments should be planned to qualify the facility, as well as to confirm the scaling capabilities of the designed SETF or ITF. Counterpart Tests (CTs) and Similar Tests (STs), should be planned, executed, and analysed (see discussion in Section 3.3.1). A scaling report related to experiments should be issued wherein the scaling distortions identified at design level, and those resulting from executing the CT or ST are discussed.

VII) The second tool for scaling analysis, aside from the ITF or the SETF, is the system’s thermal-hydraulic code. The code scalability or the code scaling capability is evaluated here. [The concept of code scalability addressing the verification process of the TH code was discussed through examples at the 2nd meeting of the SSG for issuing the S-SOAR, D’Auria, 2014. The idea is that the established scaling laws part of the equations, e.g. the distributed friction pressure drop is ‘only’ function of velocity and its equivalent diameter, and not a function of the area (scaling parameter in this case); this should be verified when the scaling parameter is changed.]

VIII) The issue of interest here is the objective established at step I). A resemblance should be evident between the thermal-hydraulic phenomena observed in the experiments and the ones expected in the objective NPP calculation. Furthermore, key parameters representative of those phenomena should be identified. Experimental data (or data derived from experiments) at different scales should exhibit same thermal-hydraulic phenomena and pertinent parameter values. The similarity of the key parameters at different scales should be demonstrated.

IX) The same processes as in the previous step shall be repeated here. However the word ‘parameter’ should be substituted by the word ‘accuracy’, or ‘error’ resulting from the comparison between the code’s calculation and the experimental data.

X) At this point in the overall procedure, background material is needed to show scaling capabilities are available from previous activities: i.e. scaling data are available and qualified, the code is qualified against scaling, and the scaling data is used to develop and qualify SETF/ITF code nodalizations. Furthermore, that NPP nodalization has been developed taking into consideration of the experience gained in modelling the phenomena that are expected for the reference transient scenario defined in step I) and in analysing the related results. Then, the activities at the present step are synthesized by the first bullet item of the macro-step D., i.e. "… performing the Kv-scaled calculation … ".

XI) Whatever is written for the step X) can be repeated here. Then, the activities at the present step are summarized by the second bullet item of the macro-step D., i.e. "…demonstrating that phenomena predicted for the NPP transient… are consistent with those in the experimental data base…".

Acceptability thresholds are verified at steps VII), VIII), IX), and X). Qualitative- and quantitative-acceptability thresholds at steps VIII), IX), and X) were proposed within the framework of the application of UMAE-CIAU to meet safety requirements, e.g. D'Auria et al., 2012, see also section 4.2. The acceptability of the requirements is determined solely by the regulating authorities. The approaches discussed in section 4.2 can be applied to define the threshold values. In the case of scaling in code verification, step VII), more stringent values for acceptability should be used than those adopted for the other steps.

4.6 The support to scaling from CFD tools

4.6.1 The CFD codes and their capabilities

When multi-dimensional effects are playing a dominant role in a safety- or a design-issue, system codes cannot be used with sufficient confidence, and 3-D CFD tools are used more and more used for investigations. However, to allow the use of CFD in licensing, important requirements are to be met, including Guidelines, V & V, and UQ. NEA/CSNI WGAMA played a significant role in the past decade in promoting the use of CFD for Nuclear Reactor Safety.

Single-phase CFD

Within the past activity of WGAMA on CFD application to NRS, there was an evaluation of the existing basis of CFD assessments, identifying gaps that need to be filled so to adequately validate CFD codes, and propose a methodology for establishing assessment matrices relevant to the NRS needs, Smith et al., 2008.

Considering only single-phase issues, most of them are related to turbulent mixing problems, including temperature mixing or the mixing of chemical components in a multi-component mixture (boron in water, hydrogen in air, etc.; no distinction is made hereafter between DBA ad BDBA conditions, the latter also including severe accidents):

- Erosion, corrosion, and deposition;
- Boron dilution;
- Mixing: Stratification/hot-leg heterogeneities;
- Heterogeneous flow distribution (e.g. in the SG inlet plenum, causing vibrations, etc.);
- BWR/ABWR lower-plenum flow;
- PTS (pressurized thermal shock);
- Induced break;
- Thermal fatigue;

– Hydrogen distribution;
– Chemical reactions/combustion/detonation;
– Special considerations for advanced reactors (including Gas-Cooled ones).

One may add the issue of a main steam-line break (MSLB) where there is mixing in the Pressure Vessel (PV) between cold water coming from the broken loop, and hotter water coming from the other loops.

In some mixing situations density differences induce buoyancy effects that have a significant influence on the mixing: Cold water may be mixed with hot water, borated water mixed with non-borated water, hydrogen with air, and so on.

All these mixing problems may be simulated with either the Reynolds Average Navier Stokes (RANS) or the Large Eddy Simulation (LES) models of turbulence, but RANS models require less CPU costs and are likely to be preferred. The choice between the various types of turbulence models may depend on the situations and some advice is given in the OECD-CSNI's Best Practice Guidelines (BPG), Mahaffy et al., 2007.

Among the mixing problems listed above, only thermal fatigue requires that low-frequency fluctuations be predicted, which almost excludes RANS approaches and gives strong added value to the Large Eddy Simulation (LES).

Uncertainty quantification is needed for CFD and should focus first on mixing problems with density effects in the steady state or in slow transients, since that would cover most envisaged applications. A review of UQ methods for single-phase CFD is in progress in the frame of a Working Group of the OECD-WGAMA.

Two-phase CFD

Two-phase CFD is much less mature than is single phase CFD, but significant progress was made in the past decade. A Writing Group of the OECD-CSNI, Bestion et al., 2006, Bestion, 2010, and Bestion et al., 2010, listed the issues in nuclear-reactor safety issues that may benefit from using of two-phase CFD. Whereas for most of these issues the existing technology already offers solutions, employing two-phase CFD may provide more accurate and/or more reliable solutions offering both a higher safety level, and better efficiency in the reactor.

Three successive European collaborative projects contributed significantly to the progress of two-phase CFD, namely the NURESIM (2006-2009), NURISP (2010-2012) and NURESAFE (2013-2015) projects. Their main results are related to the use of CFD to bubbly flow and boiling flow, including CHF investigations, to Direct Contact Condensation- and PTS-investigations.

Among the outcomes is the publication of a State of the Art in Two-phase CFD in a special issue of Multiphase Science and Technology with papers on the various 2-phase CFD approaches, Bestion et al., 2011, on adiabatic bubbly Flow, Krepper et al., 2011, on boiling bubbly flow, Koncar et al., 2011, on annular mist flow, Anglart & Carraghiaur, 2011, and on stratified flow, Lucas et al., 2011.

Due to the lesser maturity of two-phase CFD tools, the modelling will require probably several decades of R & D work before full application to design and safety problems in NPP. However, the first applications of such tools already may be envisaged, provided that a rigorous methodology of modelling and validation is applied.

The multiscale simulation

The analysis of any reactor-issue highlights the fact that different scales are involved and it is natural to investigate them with simulation tools at different scales. The multi-scale approach was presented by Bestion, 2010, and Bestion, 2012.

In a two-phase flow thermal-hydraulics analysis for nuclear applications, one can distinguish three different 3-D-simulation scales that can be classified as follows:

– CFD in porous medium: Often referred to as the "component" scale, this scale is dedicated to studies of the design, safety, and operation for reactor cores and tubular heat-exchangers (steam generators, condensers, and auxiliary exchangers). Rod- or tube-bundles are homogenized into the control volumes using the "porosity" concept. The minimum spatial-resolution is fixed by the sub-channel’s size (scale in centimeters) in the sub-channel analysis codes.

– CFD in open medium: the average scale (millimeter or less) allows us to go beyond the limits of the component scale for a finer description of the flows. It includes modelling turbulence modelling using the RANS approach, and new approaches similar to the LES may be used in some flow regimes. One can envisage a local analysis in the critical parts of the cores, steam generators, or other components including complex geometries. It is also the only scale able to predict the fluid’s temperature-field for investigating thermal shocks or thermal fatigue in the reactor’s structures.

– Direct Numerical Simulation (DNS): In the absence of any space or time averaging, the characteristic length may be less than a micrometer. This allows local simulations focusing on very small domains (e.g. containing only a few bubbles or droplets). The use of DNS will help in understanding local flow phenomena and starts by being used for developing closure relations for more macroscopic models. Besides this, it may provide basic information for the larger scales.

CFD tools may be used at different steps of a reactor thermal-hydraulic issue. They may help identifying and ranking dominant processes during a PIRT and scaling analysis, particularly when 3-D effects are supposed to play an important role. They also may be used as a support to build a nodalization (an input deck) for a system code. They may serve as complementary tools to extend the validation of system codes, and also can be employed for simulations at reactor scale provided that the scalability of the code (with its models) is sufficiently well established. In this last application, they may be chained to or coupled to a system code. These applications are presented in the following subsections.

4.6.2 The use of CFD codes in the preliminary scaling analysis

Within the preliminary scaling analysis of a reactor’s thermal-hydraulic issue, there may be several possible utilizations of CFD calculations as follows:

(1) CFD as a tool for ranking processes: For analysing thermal-hydraulic phenomena to evaluate the importance and/or ranking of specific phenomena among many of the thermal-hydraulic responses during reactor accident, and/or abnormal transients. In this type of utilization, the phenomena within scaled-down test facility may not need to be considered, but only those in the prototype reactor. Examples are the following :

   – Impact of a temperature stratification on a typical situation: One may consider situations with a break or a leak in a component (Cold leg, lower plenum…) which may be fed by a stably-stratified water layer with significant temperature gradients against another water layer. Then the flowrate – critical flow – through the break may depend on 3-D phenomena around the break upstream that control and are controlled by this stratification. CFD calculations may be useful to quantify this stratification and then to evaluate its impact on the whole thermal-hydraulic problem.

   – When investigating Pressurized Thermal Shock and the process of thermal mixing in the region between the ECCS injection and the pressure vessel. CFD can give a quantitative estimation of this mixing and estimate the extent to which the existing scaled ITF may distort this mixing.

This use of CFD requires a sufficient degree of confidence in the CFD code for the situation of interest, especially when this is directly used for a (non-scaled) reactor problem.

(2) CFD as a tool for scaling IETs or SETs: When simulation experiments are performed with either scaled IET or SET in a reduced size, under different pressure, and/or by using different fluid, the reactor system should be well scaled-down to properly represent the most important thermal-hydraulic phenomena under designated conditions that may appear during accident and/or abnormal transients of interest. Especially for the IET experiments, care should be taken to consider interaction(s) of phenomena simulated within differently scaled-down components that may have different influences from those components with different scaling laws, distortions, or concepts etc. Planning scaling thus is extremely important for considering the appropriate experimental boundary conditions to best meet the observed phenomena to those under prototypical reactor conditions. Since all the single-phase- and two-phase thermal-hydraulic phenomena are three-dimensional (3-D), two-phase CFD analysis may be helpful for designing the facility. For such preliminary scaling analyses, both the thermal-hydraulic responses in the reference reactor and in the scaled-down facility are calculated within the flow conditions of interest. If significant distortions are found, one would analyse the reasons and test other designs. Examples are the following:

- Mixing in a lower plenum in the case of a steam line break or in a boron dilution problem may depend a lot on geometric peculiarities. CFD may be used to check to what extent a scaled facility scales the mixing in this component, and an optimization of this geometry is possible through sensitivity tests (performed by CFD codes).
- When some IET component induces some distortion of a shape, singular pressure losses in complex geometries may not be respected, and CFD may be used to estimate the distortion and to optimize the geometry.

Once the experiment results are obtained by using the scaled-down facility that was constructed after the design calculations, the results can serve to validate both the one-dimensional computer codes, and the CFD codes used for single- and two-phase flow analysis within the given scale of phenomena.

4.6.3 The help of CFD codes to build input decks for the system code

One-dimensional (1-D) system codes are used for analysing reactor safety either for conservative- or best-estimate-calculations. To develop an input deck for the reactor system of interest, difficulties may be encountered on deciding how to provide some coefficients of models, and correlations used in the system code when they are very dependent on the specific geometrical design, or when they were tested and characterized in conditions far from the reactor's conditions. Such coefficients may include form-loss coefficients at rather complicated reactor structure with multi-dimensional flows under either single-phase- and two-phase-flow conditions. Since the size, flow conditions, and fluid conditions of reactor sometimes are far beyond the reach of experiments performed under the room temperature and the available (very large) test facility, the 3-D CFD flow analysis may be one possible means to estimate the flow conditions and provide the coefficients for models and correlations used in the system's code. This also is important for a quantitative evaluation of uncertainties in the results obtained by the system's code.

One example is the calibration of flow rate in the (flow rate measurement nozzle) of a BWR's feed-water line. In these conditions the Re number exceeds 10^7, and only a few facilities are adaptable for the verification test under the same Re number. The extrapolation of the characteristic of the flow-measurement nozzle obtained under a lower Re number through some method such as the one recommended by ASME is not well certified in the uncertainty evaluation. The experimental error (or uncertainty) should be strictly limited within a certain small value of 0.25%, Furuichi et al., 2014. The utilization of the CFD calculation after validating its use for extrapolating the phenomena extrapolation in the flow- measurement nozzle then is considered as one of methods to identify true uncertainties at prototypical flow conditions, once the verification test facility is unavailable. In this application, a very detailed 3-D measurement of component's shape also is needed to define the boundary conditions for the CFD calculations.

Another example is an estimation of form-loss coefficient within the core bundle. When the reactor safety analysis is performed, the reactor core needs to be modeled by using multiple channels; one of them should have a peak-power profile with which to assess the safety margin via the peak cladding temperature (PCT). Under various reactor accident situations, even for a circulation having a rather statistical nature, flow mixing may happen in the core among fuel bundles with different power profiles and magnitudes. Valid value(s) of the form-loss coefficient for cross-flow in the core is not given in many cases. Guessing the value is not appropriate, as it may occasionally cause large uncertainties in the calculated results. Since the lateral flow component is feeble, measuring it within the fuel bundle is experimentally difficult. Then, the CFD code calculations for single-phase- and two-phase-flows are suitable to identify the form-loss coefficient for cross-flow among fuel bundles (fuel rods).

Other examples on the form-loss coefficients (difficult to obtain) may include those at (or controlling): (a) the inlet nozzle of a BWR Jet pump, (b) the separator and steam dryer of BWR and PWR SG, (c) the cross-flows in the PWR's upper plenum and the BWR's lower plenum among the CRGT (control rod guide tube), (d) the bypass flow paths between the PWR hot leg and upper plenum and the PWR hot leg and downcomer (the resulting uncertainty is rather large), (e) the BWR side-entry orifice(s), (f) the flow leakage around the bottom of the BWR channel box, (g) the flow path through PWR CRGT that connects the RPV's (reactor pressure vessel's) upper plenum and upper head, (h) the pump impeller of the main circulation pump for BWRs and PWRs.

Additional examples are the CCFL coefficients expressed in terms of conduit shape at the SG inlet/hot leg inclined pipe, the SG U-tube bottom (both ends), the PWR pressurizer surge-line's nozzle at hot leg (inclined in many cases), the core's upper tie plate, the BWR side entry orifices, and the channel box bottom entry, including the leakage path.

In future, when the two-phase CFD tools will have proven their capabilities, they also may be used to estimate flooding and the CCFL limits in some particular geometries if no other source of information is available.

When selecting 1-D-, 2-D-, or 3-D-modules, some reference CFD calculations may help.

Thermal stratification in some low velocity components, such as a pressurizer, Core Make-up tank, may be competing with 3-D turbulent mixing. A preliminary reference calculation with CFD may help for selecting a 0-D-, 1-D-, or 3-D-nodalization structure with system code modules.

Natural convection phenomena originated by power exchanges between fluid and walls may be present within a component and may not be well simulated by 0-D or 1-D modules. Comparison between a 3-D CFD simulation and 0-D or 1-D simulations may give an estimation of the errors made with a coarse nodalization, and may suggest better ways of modelling using a combination of 0-D- and 1-D-modules by system TH codes.

Some system codes may model containment with a 0-D module or by a combination of several 0-D modules for each compartment. Here also, a preliminary reference calculation with CFD may help for selecting the nodalization depending on the physical situation.

The requirements for having a sufficient confidence in the CFD have to be addressed:

- CFD codes solve time-averaged or space-filtered equations that use closure relations. They need to undergo a rigorous Verification and Validation process before being used.
- Single-phase-CFD may be used for some predictions provided that some Best-Practice Guidelines are followed, and that the selected numerical and physical options are validated against the type of flow condition of interest.

Two-phase CFD has reasonable maturity for a limited number of flow situations. For many other situations, in cases of a very complex interfacial structure, no two-phase CFD code has a reliable modelling approach so far.

Many physical model options are available in two-phase CFD options and care must be taken to use a consistent set of them. When the physical model options are selected, an exhaustive validation is necessary that covers all important physical processes and can validate all important closure relations (i.e. including addressing the scaling issue).

4.6.4 Using CFD codes to complement the system code's validation

A CCFL (counter current flow limiting) may occur at the inlet of PWR SG when the reflux cooling mode is established during an accident. The amount of coolant that comes back to the reactor core through the hot leg is limited because of the forward flow of steam that came from the core. Since the amount of coolant that flows back into the core controls the core's temperature if the core is uncovered to steam, the accurate simulation of CCFL at the SG inlet is very important. When 1-D system code is used, a set of coefficients usually is required to be the input. However, the coefficients depend on the geometrical shape of the SG inlet: i.e. the hot leg bend (that connects hot leg to SG inlet plenum) and the edge shape: round, sharp or other, at the nozzle connection to the SG inlet plenum. It then is difficult to identify the CCFL coefficients experimentally because of complex two-phase 3-D flow that appears in a conduit of prototypical size. The CCFL response, furthermore, is known to depend on the pipe's size and shape.

In many cases, the guessed values are used as the CCFL coefficients with no valid evidence to support this. Then it is difficult to correctly evaluate uncertainty in the calculated results, especially for the reactor safety analyses that have no reference data. While the UPTF [see section 3.2 for suitable references] provided excellent data that may address this problem with a reactor-sized facility, the given result is not universally true because the CCFL characteristic is keyed to the channel's shape and size. Recently, the CCFL characteristic at the PWR SG inlet was obtained via CFD analyses by comparing the calculated results between prototype and SET simulations, e.g. Kinoshita et al., 2012. While the flow is expected to become highly turbulent in steam-water two-phase flow within a large-diameter pipe at up to rather high pressures, the authors just applied the k-εmodel with first-order upwind difference scheme to their CFD calculations. Although such a combination of models may provide some erroneous results, the authors obtained good results that simulate well their 1/10-size air-water two-phase flow experiment, and the full-size UPTF results.

This is an example of a trial on this type of application of CFD code to the reactor's phenomena as a complement to the system code. Further attempts may appear as computer CPU power is continuously improved.

Other examples may be listed:

- Lower plenum voiding at the end of the blowdown phase of a LBLOCA typically is a 3-D situation with a rather simple phenomenon: Kelvin-Helmholtz instability at the free surface of the liquid may entrain liquid to the break and control the remaining mass. 0-D models are validated on some existing data, but extrapolation to new geometries (the internal structures of a

Lower Plenum depending on the reactor's design) may be investigated with CFD to check the degree of generality of the modelling.

- Many thermal-hydraulic physical situations that currently are simulated with 1-D models are often validated in idealized 1-D SETs, although the reactor situation presents some 3-D effects. In this case the CFD simulation with prototypical 3-D effects may be used as a reference to compare with the 1-D prediction. This extends the validation of the 1-D module to situations with some distortions due to 3-D effects. Examples are the following:

  - Pipes or annuli with non-uniformly heating or cooling walls: the heating or cooling may not be symmetrical, leading to 2-D or 3-D effects, possibly to some internal natural-circulation phenomena. A comparison of the 1-D and 3-D simulations may or not validate the simplified 1-D treatment.
  - Core blowdown, core uncovery, and core reflooding were extensively investigated using uniformly heated rod-bundles, although some differences existed between neighboring rods. A 3-D sub-channel analysis code may be used and compared with a 1-D simulation to check the validity of the simplified 1-D treatment.

4.6.5 Using a coupled system code – CFD code

The PIRT and scaling analysis of a reactor's thermal-hydraulic transient may reveal that dominant phenomena occur at a local scale which cannot be seen by the system codes. In this case, using CFD codes with the initial- and boundary-conditions given by a system code may be a solution for scaling up to the reactor scale. This currently is envisaged for many issues governed by single-phase turbulent mixing in a complex 3-D geometry, such as MSLB, boron dilution, and PTS. A coupling of system codes with the CFD code also is a way to solve the problem. However, this raises several questions:

- The CFD tool must be demonstrated to be applicable to the process of interest,
- The CFD tool must be validated against appropriate data, relative to the dominant processes,
- The selection of numerical options and of a turbulence model should be justified according to the situation of interest using existing Best Practice Guidelines,
- The numerical errors must be controlled (estimated to be within reasonable limits). This includes requirements on the numerical scheme and on the nodalization or meshing.
- The impact of boundary conditions given by a much coarser simulation tool (namely a 1D model) on the process of interest should be evaluated, and should not greatly affect the accuracy.
- The initial conditions in the CFD domain of simulation must be known with sufficient accuracy to avoid too much uncertainty during the time period of interest.
- The uncertainty of the predictions should be evaluated.

Single-phase CFD tools exist that now are mature enough to be applied with some confidence to some safety issues, but the cost of CPU necessarily is high in reactor applications. However, the last requirement on uncertainty quantification still is the main difficulty since UQ methodologies are not mature enough for CFD application to nuclear safety, and one may expect an even higher CPU cost.
The verification and validation of a coupled SYS TH code + CFD code simulation tool also raises questions:

- Both the system's code and the CFD code must be verified properly and validated in the domain of application.
- The coupling itself must be verified.

– The validation of the coupled tools requires that the IET test has sufficient measurements to validate both the system code for the whole system, and the CFD code with local measurements in the domain of application.

Two-phase CFD tools are less mature than single-phase tools but a few applications to nuclear safety may be envisaged in the short- and medium-term. Significant progress was obtained in some two-phase PTS scenarios, Coste et al., 2011, Coste & Merigoux, 2014. However requirements on V & V and Uncertainty Quantification (UQ) induce even more work than in single-phase scenario, and the CPU cost is even larger.

4.7 Key Findings

1) PIRT and scaling both have a key role for code development, its V & V and application, including an uncertainty evaluation. All identified dominant phenomena should be properly modeled in the code and validated (i.e. considering scaling).

2) Traceable procedures are recommended to show the role of scaling within code development, V & V, and uncertainty-evaluation processes.

3) The mature SYS TH codes, which followed traceable procedures for development and are extensively validated and verified, are powerful tools to assist PIRT and scaling analyses. Scaling methods, such as H2TS, FSA, DSS and SYS-TH codes are independent tools that can be consolidated mutually. Scaling methods provide requirements for the code model capabilities, for the V & V, for the UQ and for the code's application methodology. In return, code application may confirm, to a certain extent the adequacy of the PIRT and scaling results.

4) SYS TH codes are used to determine, or to verify scaling of the test facility using the triad method with a comparison of code simulations of the prototype, an ideal-scaled model, and the scaled experiment.

5) Kv-scaled procedures are essential for validating nodalizations, i.e. key computational tools indispensable for applying the SYS TH codes.

6) For most reactor issues, applying the SYS TH code often is the best way and the only way to accomplish up-scaling from IET to the reactor by compensating for the scale distortions of the IETs. However, there are requirements to demonstrate the up-scaling capabilities that ideally should include the following:

   o The code has been validated on the transients of interest performed in scaled IETs that represent the main phenomena of the transient as identified in a PIRT, and also predict well, both qualitatively and quantitatively, the main phenomena.

   o The code has been validated on the transients of interest performed in several scaled IETs at different scales (counterpart tests), and the code predicts the effect of scaling or its absence.

   o Within the process of developing the code, it has been proved that closure laws have a good up-scaling capability by the validation of all important phenomena at both local- or component-scale against several SETs at different scales, thus covering as much as possible of the prototypical thermal hydraulic range of interest.

   o Since scaled IETs are necessarily characterized by some scale distortions, the code should be able to predict correctly the distorted phenomena. This may require validating the distorted (in IETs) phenomena in non-distorted SETs.

   o The code is used in reactor simulation with the same numerical schemes and numerical options as were used for its validation on SETs and IETs.

- o The code is used in reactor simulation with the same set of equations and closure relations and the same empirical constants as were used for validation analyses based on SETs and IETs.
- o The code is used in reactor simulations with a nodalization and a time step as close as possible to those used for validations on SETs and IETs relative to the physical situations of interest, and following all recommendations on the best nodalization and time steps that were derived from validation studies, and that may be given in User's Guidelines.

7) CFD codes have the current role to support results of analyses performed by SYS TH codes.

8) CFD codes may be used in the near future in the same way as SYS TH codes, first for some single-phase issues wherein local 3-D effects play a dominant role. This may be the case for a few mixing problems (boron dilution, PTS, MSLB and thermal fatigue). Requirements about code development, V & V, and UQ have to be satisfied in the same way as for SYS TH code. Two-phase CFD may follow in the medium term.

5. CONCLUSIONS

A report on the current status of scaling activity was requested by the WGAMA technical group of the NEA/CSNI during its yearly meeting in 2012. A group of experts proposed a State of the Art Report on the activity "Review of Scaling (S-SOAR)" The proposal was accepted and the related research was undertaken by over a dozen scientists during a two-years-plus period. The present document is the response to the request. The objectives proposed (section 1.2) are assumed to be consistent with the request for the activity, and the expectations shall be evaluated from the text below.

A large technological base related to scaling is available at both NEA/CSNI and in the scientific community. The scaling bases have been set by CSNI since the 80's (see Chapter 1 for details) when the SOAR on TECC was issued. It was followed by the SETF- and ITF-CCVM reports, respectively:

– The SOAR on TECC describes the origin of system thermal-hydraulics and of the framework for developing the SYS TH codes;
– The SETF CCVM and the ITF CCVM deal with the systematic exploration of the thermal-hydraulics space, and envisage the requirements for demonstrating the capabilities of those codes; viz., phenomena identification during accidents in water-cooled reactors and their characterization.

Scaling has a central role, and is a key word in the SOAR on TECC, and in the SETF and ITF CCVM reports. Among the other things, the importance of Counterpart Tests and the presence of unavoidable scaling distortions in experiments were stated as established knowledge. Furthermore, 'scaling consistency and needs' constituted the topic of a CSNI follow-up document for the SETF CCVM report. In this case, needs and consistency were systematically evaluated in relation to each of the identified phenomena in transient thermal-hydraulics. In addition, the activities denominated as International Standard Problems, (ISPs, a couple of dozen ISPs on thermal-hydraulics were completed with wide international participation), and always had a connection to, and conclusions related to scaling.

The research on scaling has been active for a long time. In 1998, the status of scaling was presented in a special issue of the Journal of Nuclear Engineering and Design (J NED).The scaling distortions and a method to provide a hierarchy of scaling factors were established. Hundreds of papers have been published in relation to scaling after that special issue (more than 500 papers are listed as references in the current report). Varieties of scaling achievements were obtained and are reviewed within the S-SOAR. Attempts also have been made by individual authors since that time to reach a consensus in relation to the meaning of the scaling and associated terms, and to the ways to address scaling. Those attempts are valuable from a scientific standpoint and suitable to improve our understanding and communication; however, an international consensus on scaling was not achieved.

The motivation behind the present document is to report about numerous international scaling activities undertaken (i.e. by CSNI) and about any consensus reached in relation to the outcomes of scaling activities. Furthermore, the applicability of thermal-hydraulic findings, and methods, including SYS TH codes, to evaluating the safety of nuclear power plants with the BEPU (Best Estimate Plus Uncertainty) approach, made it desirable, or even mandatory, to achieve a common understanding of scaling issues, and namely, of the scalability of computational methods.

The focus of this S-SOAR is on addressing the transient performance of NPPs, i.e. accident analysis, where two-phase flows are encountered. In this context, the full scale- or prototype-data usually are not available due to their cost and their feasibility. Thus, a reduction in scale for geometry, pressure, and power, or combination of them, are unavoidable and are the origin of the scaling issue.

The scaling knowledge gathered so far includes the design, construction and execution of experiments in ITF and SETF. It also encompasses the technology for developing and qualifying models and the SYS TH codes that constitutes the foundation for undertaking scaling studies. The scaling analysis identifies dominant processes and plays an important role for the SYS TH codes, namely, in the development, in a comprehensive V & V processes, and in the application involving the uncertainty evaluation. Dominant processes must be modeled, validated, and the uncertainty of the model must be quantified.

The recognition of the impressive amount of experimental- and theoretical-research dealing with scaling within nuclear thermal-hydraulics should be seen as the first step for developing a common understanding. Suitable approaches for scaling have been developed and applied. Extrapolating the measured data from experiments of reduced scale to the conditions of the prototype is only for reference; it is not suitable for safety evaluation because scaling distortions are embedded in the experimental data. Therefore, extrapolation is not acceptable when the target is the accident analysis of an NPP. Here, the scaling distortion refers to the difference between the prototype and the scaled value of any plant parameter of interest.

Nevertheless, the experimental data measured in systems of reduced scale are vital for identifying and characterizing (and understanding) the transient thermal-hydraulic phenomena expected in the NPP. An important role of experimental data lies in the assessment of SYS TH codes, as well as of any computational tool necessary for their application (notably the nodalization): the validation process usually includes a review of code models, the assumptions, the adopted method of numerical solution and the nodalization. These steps should include a consideration of scale effects of the model and the data. A validation process is carried out at both a qualitative and a quantitative level. It may use thresholds of acceptability for the comparison between the measured and calculated data.

The validation process must take into account that a big difference between a measured and a computed parameter may be unacceptable in some situations and acceptable in others. For instance, when simulating a LOCA (as shown from the analysis of the UPTF experiments) a large error on condensation at an ECCS port may be acceptable since it does not have a big impact upon PCT; however, a similar large error is not acceptable when the target of the analysis is the two-phase PTS scenario. Definitely, how good is good enough depends on the required accuracy for the FoM in the reactor calculation, and not in the validation. A proper validation is based on a suitable number of datasets derived from differently scaled test facilities, either the SETF or the IETF, wherever available.

A solution for addressing the scaling issue is needed when evaluating the safety of nuclear reactors. Developing a new scaling method or choosing a suitable existing scaling method for a specific experiment is needed to minimize and justify scaling distortions. In the BEPU approach (i.e. application of Best Estimate codes supplemented by uncertainty evaluation) the scaling issue must be resolved because it is a specific source of uncertainty, since most of the phenomena uncertainties are estimated with data from the facilities of reduced scale. Some additional observations are given here:

a) In most cases, the possibility of extrapolating simply from experimental data is limited due to some possible distorted phenomena in the scaled facility. In any case, such direct extrapolation would require caution in considering the facility's scaling limits.
b) The extrapolation of calculated results, e.g. starting from data derived by a small-scale nodalization (whose results can be compared with the experimental data) and constructing nodalizations of larger and larger node size, is not relevant due to both possible distorted phenomena in the scaled facility, and the scaling limitations of the system code. This is true

when a small-scale nodalization either is a nodalization of a small-scale test facility or a nodalization with small meshes.

c) A technological solution to obtain reliable predictions of a reactor parameter of a particular phenomenon is possible when IET experimental data at different scales are available and verified, and when the accuracy of code predictions is not (significantly) affected by the scale of the facility.

In view of the basis of the approach, the SYS TH codes and models can be considered as the repository of current knowledge and expertise in nuclear thermal-hydraulics, including scaling information. The role of Computational Fluid Dynamics (CFD) codes also is reviewed in this report. The maturity of the CFD code for transient two-phase system analyses had not been attained at the time of the S-SOAR. Confirmatory- or support-analyses are expected to be performed by CFD codes first for single-phase situations where three-dimensional effects are identified as important, e.g. mixing problems in the reactor's circuit or in the containment.

The following are the main conclusions from S-SOAR:

1. The information in scaling studies, namely the experimental database, is available for most reactor types but has not been fully exploited.
2. Scaling methods and models are available for specific targets or objectives. The application to a generic objective may suffer from limitations of these methods.
3. Many non-dimensional scaling groups are derived in scaling methods and models: the hierarchy of these groups is important for applying scaling methods.
4. Distortions cannot be avoided in any reduced scale experiment where transient two-phase flow is involved. Even in the case of single-phase conditions, phenomena like stratification and entrance effect may induce distortions in scaling, particularly in the case when passive systems are involved.
5. The impact of scaling distortions upon the performance predicted for any reference system, prototype, or reactor, remains difficult to quantify.
6. Data from scaled experiments cannot be directly extrapolated to the reactor in most cases dealing with two-phase flow.
7. Use of a suitable existing scaling method or development of new one for a specific experiment is essential in minimizing scaling distortions.
8. The use of a well-validated and verified SYS TH code can support any scaling analysis, including checking the scaling hierarchy, evaluating the impact of scale distortions and correcting the distortions in reactor applications. For a safety determination of NPP, the application of SYS TH codes can support, but not replace the formal scaling analysis, and is the best tool for up-scaling to the reactor transient of interest after the two following requirements are met (i.e. items 9 and 10 below).
9. Uncertainty from scaling should be accounted for in the overall uncertainty when SYS TH code is used in predicting thermal-hydraulic phenomena in NPP accident scenarios.
10. Accurate evaluations of scaling uncertainty in the validation results, model correlations, numerical schemes, and nodalization are needed to meet the requirements of nuclear reactor safety.

5.1 Key findings

1) Scaling activities and analyses can be performed in a wide variety of contexts having different targets, where examples of contexts are a) the design of a facility, the design of an experiment, code-design requirements, derivation of a correlation, assessing a model; and, b) examples of targets are to reduce the cost of construction of a loop whilst still expecting valuable information related to phenomena, to demonstrate the acceptability of a correlation, and, to develop a new scaling approach.

2) The demonstration of an acceptable similarity between the prototype and the model constitutes one core problem in scaling. A variety of approaches and methods exist to meet this aim (Chapters 2, 3 and 4, especially Chapter 3).

3) To demonstrate an acceptable similarity between the prototype and the model, the industry is searching for better methods and procedures. In this case, other than the obvious consideration that any scaling method has its pros and cons, one may note that each application of scaling methods is unique. This is true in terms of the objective of the application, including the complexity of the system, and the available budget. Therefore, a deep understanding of the application and of all the phenomena involved therein is required to select the most suitable scaling approach. The issue connected with the meaning of the words 'qualified enough' (i.e. how good is good enough?) is beyond the scope of the activities defined by the present S-SOAR (see section 5.2).

4) All analytical methods, empirical correlations, code calculations, and experiments (namely the experimental data) are useful in supporting the PIRT and subsequent scaling analyses.

5) In transient two-phase flow conditions, the experimental data plays the key role in validating the results of analytical models. However, the experimental data obtained from the ITF or SETF should not be extrapolated directly to predict prototype conditions since it involves unavoidable distortions.

6) The research associated with large 'reduced-scale' ITF, or SETFs built in the past cost hundreds of millions of US dollars. Therefore, research investments were driven by nuclear-safety requirements. The finding here, perhaps well established, is that the size and the key features of the facility (e.g. its maximum power) are related to available funds.

7) Each experimental facility, either an ITF or SETF, may produce valuable data suitable for code validation. The scaling of the full-pressure, full-height, and full-linear-power of the core, targeting the one-to-one scale for simulating complex transients in real time, is recognized as the best practice for the design of the ITF and SETF. A dozen ITFs with these characteristics have been used in the last two or three decades to explore the phenomena that are the basis of the current system's thermal-hydraulics.

8) Noticeably, a group of facilities was designed and built following the 'reduced-height' scaling approach (sometimes referred as Ishii scaling). The approach of reduced height brings the benefits of cost savings (compared to a full-height facility having the same volume and power), and the advantage of larger flow areas. The latter benefits the simulation of two-dimensional and three-dimensional effects without additional costs.

    a. In cases where the key target of an experimental research is to simulate Peak Cladding Temperature (PCT), a safety-relevant parameter for NPPs, it is important to keep linear power and pressure as close as possible to the prototypic conditions. Because of the relationship between velocities of two-phase flow, void fraction, heat flux, and the fuel rod surface temperatures, the 'reduced-height' scaling implies an (undue) assumption when designing linear power for an experiment. This is not required at full height (thus the word 'undue' in previous sentence).

b. In the past, in relation to experimental programs based on reduced-scale test facilities (noticeably including LOFT and ROSA-III), the measured data were interpreted according to the knowledge on scaling at the time when the experiments were performed.

c. Definitely, full height – full pressure-scaled IETs and SETs are highly recommended to be part of the validation matrix of system codes since this is the only way to preserve all non-dimensional numbers controlling the flow regime and heat- transfer regimes in Core (and the SG for PWRs). Such facilities also use prototypical wall-heat flux and respect wall temperature, preserving similarity with respect to CHF occurrence and rewetting, which are major phenomena in many transients of interest.

d. Full-height scaling appears of the outmost importance for designing models simulating natural-circulation prototype systems.

9) Counterpart Tests (CT) have been performed in the past and are highly valuable to verify the scaling effects. They are a partial validation of the scaling analysis that led to the design of the test facility.

10) The availability of UPTF data identified distortions that could not be captured by scaling because one of the prerequisites of scaling is that the two-phase distribution is preserved. This specifically is true for flooding and CCFL in upper plenum, hot leg, and for CCFL breakdown in the downcomer during a refill.

11) The validation of a code includes qualifying the numerical models and the nodalization. This is not the main subject for the present S-SOAR. However, part of the process of qualifying the calculation, e.g. the so-called 'Kv-Scaled' method is connected directly to the topic of the S-SOAR. The Kv-Scaled method is based on comparing the NPP-calculated results with the data measured in the ITF. A scaling method is adopted (i.e. any scaling method can be adopted in this context) and conclusions are drawn based on the acceptability of distortions between the calculated and the measured data.

12) To prove the scale-up capability of a code applied to a reactor transient, an ideal list of best-practice requirements would include the following :

a. The code has been validated on the transients of interest performed in scaled IETs that represent the dominant phenomena of the transient, as identified in a PIRT. It should predict well, both qualitatively and quantitatively, the dominant phenomena.

b. The code has been validated on the transients of interest performed in several scaled IETs at different scales (counterpart tests), and the code predicts the scale effect or its absence.

c. The code has proven that closure laws have a good up-scaling capability by validating all important phenomena at local scale or a component scale against several SETs at multiple scales.

d. The code has proven that the closure laws' validation domain covers the entire prototypical thermal-hydraulic range of interest.

e. Since scaled IETs necessarily have some scale distortions, the code should be able to predict correctly the distorted phenomena. This may require a validation in non-distorted SETs of the phenomena distorted in the IETs.

f. The code is used in a reactor simulation with the same numerical schemes and numerical options as were used for validation on SETs and IETs.

g. The code is used in reactor simulation with the same set of equations and closure relations as were used for validation with SETs and IETs. However, form-loss coefficients and CCFLs are examples of parameters that depend upon geometry.

h. The code is used in reactor simulation with a nodalization and a time step as close as possible to those used for validations on SETs and IETs relative to the physical situation of interest, and following all recommendations on the best nodalization and time steps that were derived from validation studies, and that may be given in the User's Guidelines.

13) The importance given to scaling in Uncertainty Quantification can be derived from the role of scaling inside the CSAU-, GRS-, and UMAE/CIAU-procedures (Chapter 4).

14) Scaling roadmaps were discussed in relation to the design of an experimental facility, and to assessing nuclear reactor safety. In the former case, a roadmap based on the DSS method is presented as an example. In the latter case, two roadmaps are proposed, dealing with how scaling is involved in reviewing NPP safety: The scaling database available to the scientific community is exploited within the framework of those roadmaps.

15) All steps of the methodology for predicting a reactor transient require a high level of expertise. PIRT requires a good background on the phenomenology. Scaling requires a good control of the various ways to write equations from basic principles. Although H2TS, FSA, and DSS are important steps in formalizing the procedure, no method is sufficiently systematic that the application is free of error, and applying scaling methods also may have a user effect. Code application requires a good knowledge of capabilities and limitations of the codes. The use of codes cannot, in any case, be an alternative to having a physical understanding of the phenomenology. However, codes may be useful tools at every step; sensitivity calculations may help in the PIRT exercise, pre-calculations of IETs may check the scaling rules, scaling-up to reactor scale may be better done through a code, provided that the requirements for scalability are met.

5.2 Recommendations

Recommendations are provided for planning future activities. Based on the key findings they are summarized here without any prioritization.

– To resolve a safety issue related to a postulated reactor accident, the most reliable approach should combine the use of PIRT analysis (where Identification is more important than Ranking), scaling analysis, analysis of a wide SET- and IET-experimental data-base, and the use of a system code in a BEPU approach. In some cases, a multi-scale simulation using CFD tools in addition to SYS TH codes may give better insights into local 3-D phenomena.

– The capability of SYS TH codes to predict transient scenarios in facilities of different scales is needed to evaluate the safety of water-cooled nuclear reactors. The recommendation is to include the scalability requirements in SYS TH code Validation. The counterpart tests will also be important asset for validating the scalability of the codes.

– The database of existing SETF and ITF CCVM (see Chapter 4 for references), should be extended to include possibly data related to advanced reactors (including those using passive safety-systems), radial transfers due to diffusion, dispersion of momentum and energy, and cross flows in the core.

– There is a need for well-instrumented tests for validating CFD codes for the water- cooled reactor in relation to mixing problems, such as boron dilution, MSLB, PTS, thermal fatigue, or mixing with buoyancy effects in some ‘passive’ systems, to be considered in the general TH validation matrices. CFD codes first must be validated on single-phase tests at different scales.

– There is a need to identify a qualitative and quantitative framework (precision targets) to judge the quality of a scaling approach. This step is connected with the acceptance criterion for scaling distortions, and with the quantification of uncertainty due to scaling.

- Full-height scaling and suitable flow areas (and volume size) are recommended for experimental simulation of passive system wherein the important phenomena are the boiling- and condensation-processes, and buoyancy effect due to density change. Full height will provide an accurate characterization of phenomena, such as natural circulation and related stability.
- Specific scaling-related training is worthwhile in a number of contexts. On both the industry and regulatory sides, the good training and education of safety analysts should include, in addition to basic single-phase and two-phase thermal-hydraulics, advanced topics of scaling techniques, identification of the dominant phenomena of major transients, code V & V and UQ requirements, and code- scalability requirements.
- Revisiting systematically the scalability of system codes at the basic level of each closure law may be a good exercise for training new code users, so to improve the understanding of code scaling uncertainty and to improve the code's documentation.
- Multi-scale analysis using several numerical tools at different scales will help in future to provide more accurate and reliable solutions to reactor issues. This approach requires first that the capabilities and limitations of 3-D two-phase flow calculation (CFD) methods for flows relevant to an NPP are well identified. The simulation capability of details of local phenomena aiming for a replica of the phenomena must be improved. Up-scaling methods for modelling should be developed to use small-scale simulations for improving the closure laws used in SYS TH codes. The CFD tools also should follow an appropriate process of code validation to prove their capability for extrapolation to the NPP-prototype phenomena.

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LIST OF SYMBOLS

A	cross section area of a duct
A_i	flow area scale in *i*-th component (-)
a	equivalent flow area (m^2)
a_R	area scale (-)
a_s	solid structure cross-section area (m^2)
C_p	specific heat (J/kg K)
C_s	pressure wave (sound) celerity
C_α	void wave celerity
D	diameter (m)
D	scaling distortion ratio (-)
D_h	hydraulic diameter
D_R	rod diameter scale (-)
d	diameter (m)
d_R	diameter scale (-)
F_{ik}	interfacial momentum transfer to phase k per unit volume (N/m^3)
f_w	wall friction factor (-)
g	gravity acceleration (m/s^2)
g_i	gravity acceleration vector components (m/s^2)
g_R	gravity or acceleration scale (-)
H	head rise (m, ft)
H_k	time and/or space averaged specific enthalpy of phase k (J/kg)
h	specific enthalpy (J/kg)
h	heat transfer coefficient (W/m^2K)
i_{fg}	latent heat of vaporization (J/kg)
j	superficial velocity (m/s)
j_i	flux of a property (mass, momenturm, energy)
J_k	superficial velocity of phase k (m/s^2)
K	form loss coefficient (-)
k	thermal conductivity (W/mK)
L	spatial scale, length (m)
L_i	axial length scale in *i*-th component (-)
l	length (m)
l_R	length scale (-)

$\dot{M}$	mass flow rate (kg/s)
M_o	initial coolant mass inventory (kg)
$\dot{m}_R$	scale of mass flow rate (-)
N_s	specific speed
n_R	scale of number of rods (-)
P	time and/or space averaged mixture pressure (Pa)
p	pressure (Pa)
p_0	initial pressure of the primary system (Pa)
p_s	secondary system pressure (Pa)
Q	core power (W)
Q_f	volumetric flow rate (m^3/s)
Q_h	heat loss (W)
Q_{ik}	interface-to-phase k interfacial energy transfer rate per unit volume (W/m^3)
Q_s	heat source number (-)
q	flow rate (m^3/h, l/s, l/min, m^3/min, US gpm, British gpm) at Best Efficiency Point (BEP)
q_j	conduction heat flux vector components (W/m^2)
q_{kj}	conduction heat flux vector components in phase k(W/m^2)
q_{kj}^t	turbulent diffusion vector components in phase k(W/m^2)
q_{Ro}	core power scale (-)
q_R''	heat flux scale (-)
q'''	volumetric heat generation (W/m^3)
q_R'''	scale of power to volume ratio (-)
q_{wk}	wall heat flux to phase k (W/m^2)
S	slip ratio (-)
S_R	slip ratio (-)
T_{CL}	cold leg temeprature (K)
T_{HL}	hot leg temeprature (K)
T^*	Time ratio number (-)
T_R	temperature ratio (-)
T_{sat}	saturation temperature (K)
t	time (s)
t_R	time scale (-)
u	velocity (m/s)
u	specific internal energy (J/kg)
u_{gj}	drift velocity (m/s)
u_R	velocity scale (-)
V	volume (m^3)
V_k	time and/or space averaged velocity of phase k (m/s)
V_R	volume scale (-)
v_i	velocity vector components (m/s)

v_{ki} velocity vector components for phase k(m/s)
x quality (-)

Non-dimensional numbers

A_{ik}^{+} dimensionless area for a transfer process between constituents *i* and *k* (-)
B_i Biot number (-)$Bi = \frac{hL}{\lambda}$
Bo Bond number (Eötwos)$Bo = \frac{g\Delta\rho DL^2}{\sigma}$
D^* dimensionless diameter (-)
F friction number (-)
F_d pump characteristic number (-)
Fr Froude number$Fr = \frac{v}{\sqrt{gD}}$
j_{ik}^{+} dimensionless flux of a property between constituents *i* and *k* (-)
$J_k{}^*$ non-dimensional superficial velocity of phase k (m/s^2) $J_k{}^* = \frac{\sqrt{\rho_k}\,J_k}{\sqrt{g\Delta\rho D}}$
K_k Kutateladze number for phase k$K_k = \frac{\sqrt{\rho_k}\,J_k}{(g\Delta\rho\sigma)^{1/4}}$
Q_{fi}^{+} dimensionless volumetric flow rate of constituent *i* (-)
R Richardson number (-)
Re Reynolds number$Re = \frac{vD}{\nu}$
St modified Stanton number (-)
V_i^{+} dimensionless volume scale of constituent *i* (-)
*W*e Weber number$We = \frac{\rho\Delta V^2 L}{\sigma}$

Greek symbols

α volumetric concentration (-)
α_k local time fraction of phase k (in 3-D equations) or time averaged volume fraction of phase k (in 1D equation)
α_R void ratio (-)
α_s thermal diffusivity (m^2/s)
β volumetric thermal expansion coefficient (K^{-1})
ΔH_d pump head (m)
Δi_{sub} inlet subcooling in enthalpy (J/kg)
Δi_{subR} scale of inlet subcooling in enthalpy (-)
Δp pressure drop (Pa)
$\Delta\rho$ density differene between liquid and vapor (kg/m^3)
$\Delta\mu$ difference of dynamic visocity between liquid and vapor (kg/m s)
ΔT temperature rise (K)
ΔT_R scale of temperature change (-)
ΔT_{subR} scale of inlet subcooling in temeprature (-)
δ conduction depth (m)
Γ_{ik} interface-to-phase k interfacial mass transfer rate per unit volume (kg/m^3/s)

χ_w	wall perimeter
χ_f	fluid characteristic (or indicator) function
χ_k	phase characteristic (or indicator) function
Π_i	characteristic time ratio (-)
Π_{ik}	characteristic time ratio for a transfer process between constituents *i* and *k* (-)
λ	thermal conductivity (W/mK)
π	similarity parameters (-)
ψ	a property (mass, momentum, energy)
ψ_i^+	dimensionless property of constituent *i* (-)
ϕ	Porosity
μ	Dynamic viscosity (kg/m s)
μ_k	Dynamic viscosity of phase k (kg/s/m)
ν_k	Kinematic viscosity of phase k (m^2/s)
ω	pump shaft rotational speed (rpm)
ω	Specific frequency (s^{-1})
ρ	Density (kg/m^3)
ρ_k	Density of phase k (kg/m^3)
σ	Density of phase k (kg/m^3)
σ	Surface tension (N/m)
σ_{ij}	Stress tensor components (Pa)
τ	Temporal scale (s)
τ_{wk}	Wall friction force applied to phase k (Pa)
ζ	Wetted perimeter (m)

Subscripts

0	initial
C	constituent
CL	cold leg
CV	control volume
CP	phase of constituent *C*
CPG	geometrical configuration of phase *P* of constituent *C*
f	fluid, saturated liquid
g	saturated vapor
HL	hot leg
i	constituent i
i	i-th component
k	constituent k, phase k
m	model (test facility)
o	reference point/component, reactor core
p	prototype
R	ratio of the value of a model to that of the prototype

s	solid
sat	saturation
x	x-direction
y	y-direction
w	wall

GLOSSARY

The Glossary takes benefit of a diagram showing the role of scaling in nuclear thermal-hydraulics. The diagram is given in the Fig. G-1 below and includes some of the key terms listed in the Glossary.

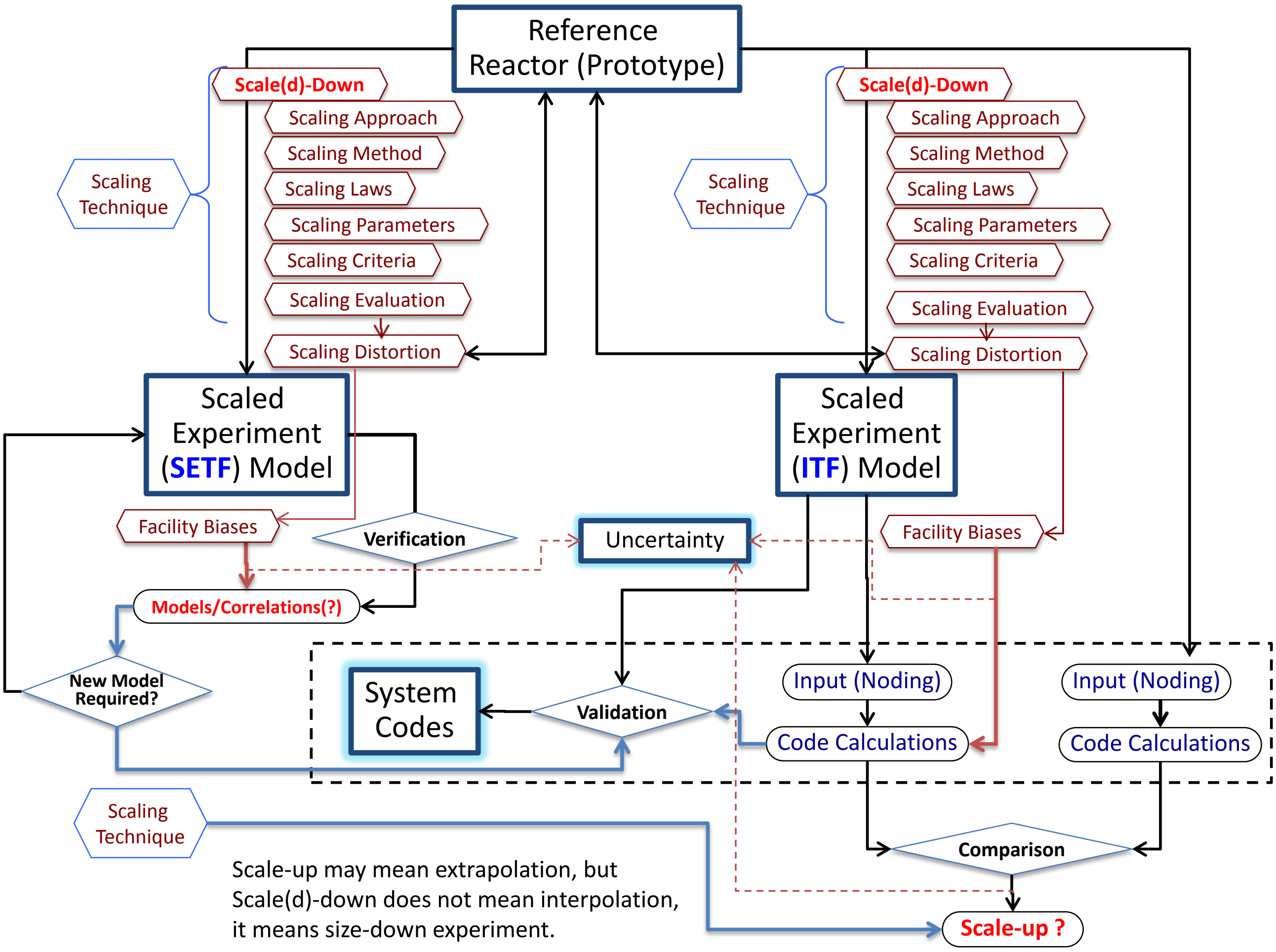

Fig. G-1: Role of scaling in nuclear thermal-hydraulics to support the identification of terms in the Glossary.

ADDRESSING THE SCALING ISSUE
See Scaling

BEST ESTIMATE CODE AND APPROACH
A best estimate code (approach) uses more realistic information about plant behaviors and phenomena for calculation and analysis. It usually combines with uncertainty analysis to provide realistic margin between the results and acceptable criteria, Gupta et al., 2012.

CODE
The word 'code' is interpreted as numerical or computer code. The numerical code is an ensemble of equations and correlations suitable to calculate the transient performance of thermal-hydraulic systems, including NPP and involving the evolution of two-phase mixtures.

CODE SCALABILITY (code scaling-up capability)
Codes which are validated on scaled (lower scale to full scale) SETs and IETs may have the capability to predict the phenomena of interest at facility of other scale or reactor condition. This capability is called Code Scalability (see chapter 4 and section 4.3 for more details).

CODE SCALABILITY LIMITATION
They are the conditions that the requirements of code scalability are not met. Please reference Code Scalability.

CODE SCALING DOWN CAPABILITY
Codes which are validated on higher scale SETs and IETs may have the capability to predict the phenomena in facility of lower scale. Some limitations such as the range of operating condition may apply. This capability is part of Code Scalability.

CODE SCALING-UP CAPABILITY
Codes which are validated on lower scale SETs and IETs may have the capability to predict the phenomena in facility of higher scale or full scale prototype. Some limitations such as the range of operating condition may apply. This capability is part of Code Scalability.

COMPLEMENTARY TEST
It refers to a test combing ITF and SETF to cover different aspects relevant of an accident scenario.

CONSERVATIVE CODE AND APPROACH
A conservative code (approach) is to the opposite of best estimate code (approach). In conservative code (approach), all information used is based on conservative assumptions regardless of plant behavior and phenomena. The approach does not show margin between the actual response and the estimated response for operational flexibility. This approach was introduced in 1970s due to limited capability of modelling, Gupta et al., 2012.

COUNTERPART TEST
Experiments performed in differently scaled models, i.e. models designed and constructed preferably following an assigned set of scaling factors and differing among each other for the value of parameters characterizing either the geometry, or the pressure, or the power, according to a set of BIC resulting from the application of suitable scaling factors. In the case of CT one reference experiment should exist and distortions in BIC values related to the ideal scaled values should be either non-existent or minimal and explained (consequences of distortions quantified).

DAUGHTER TEST
It refers to a special counterpart test using full scale test data as reference to compare with results from scaled-down experiments on the same phenomena/processes (see section 3.3.3 for details). It is suggested that daughter test can be called 'Special Counterpart Test'.

DESIGN SCALING LAW
Mathematical relationships that relate design variables of the original and scaled systems (see also Scaling Law).

DISTORTION
See scaling distortion

EVALUATION MODEL
A calculation framework for evaluating the behavior of the reactor system during postulated Chapter 15 events, USNRC, 2000, which includes one or more computer programs and all other information needed for use in the target application.

FACILITY BIASES
Test results given from a test facility with some scaling distortion, which have some discrepancies from those "expected" in the reference plant response under the same boundary conditions

FIRST PRINCIPLE COMPUTER CODE
The computer codes with numerical calculations which are started from established laws of physics and have no assumptions using empirical models and fitting parameters. The definition is an ideology in the domain of thermal hydraulics calculations; no computer codes are qualified so far.

IDEAL SCALING
It refers to an idealistic (or virtual) design of a test facility that is ideally scaled down from a prototype by using a selected scaling method including associated implied approximations and compromises. The dimension or conditions are derived mathematically from the scaling method, without adding any modifications by the applicants trying to improve distortions. It does not consider any manufacturing limitations either - e.g. heat losses, nuclear fuels, piping configurations, materials... Instead, it uses the scaled nuclear fuel geometry and properties, the scaled heat losses, the scaled piping arrangements from the scaling method. Very often the ideal scaling is not possible due to manufacturing limitations or due to contradicting scaling criteria. For example when scaling down a pipe, one cannot respect the ratio of fluid to metal heat capacity, the time scale for heat release from wall to fluid, the ratio of heat exchange area and fluid volume. Usually manufacturing limitations result in too much solid volume, non-prototypical heat losses, too much wall thermal inertia (see Section 4.1.5 regarding the verification of scaling laws in Ransom's paper, Ransom et al., 1998).

IDEAL APPLICATION OF A SELECTED SCALING METHOD
The implementation of scaling using Ideal Scaling concept.

IDEAL FACILITY
Contrary to real (actual) facility, the facility designed by ideal scaling is called ideal or virtual facility.

MODEL
Scaled-down (experimental) system designed and operated in order to simulate the performance of the prototype.

NODALIZATION
The set of input data developed and needed for a SYS TH code calculation. The nodalization is the results of a brainstorming process performed by the code users where the facility to be modeled and the features of the specific-concerned code play a role.

NODING
See Nodalization

PROTOTYPE
The concerned nuclear system, water cooled nuclear fission reactor, whose transient performance constitutes is the target for the scaling studies.

REAL (ACTUAL) FACILITY
Contrary to ideal (or virtual) facility, the real (actual) facility is the facility including all distortions from the scaling method and the construction deviations.

RCS and PCV SCALED DOWN FACILITIES
Scaled-down facilities are used to characterize the thermal hydraulic behavior of a nuclear power plant by investigating the local/component and overall/system phenomena. The phenomena are to describe the interaction between atmosphere, structure/components and the fluid. Two main categories of tests, separate effect test facility (SET) and integral test facility (ITF), are designed to accomplish target functions of RCS and PCV.

A Separate Effect Test Facility (SETF) is an experimental test facility designed to investigate: 1) the reactor component behavior (SETF-Component test) by characterizing the component responses that are typical of the design function; 2) the local phenomena (SETF-Basics test) in order to validate closure relations. One phenomenon or several combined phenomena can be investigated in one SET.

An Integral Test Facility (ITF) is a scaled-down test facility designed to investigate: 1) the overall system behaviors and the related phenomena and processes; 2) the interaction of two or more components; and 3) the local phenomena those are typical of the overall system design target function.

In the framework of the new advanced reactor, some designs are characterized by a mitigation strategy based on RCS-PCV coupled thermal hydraulic behavior. Integrated integral test facility (IITF) and coupled integral test facility (CITF) belong to this general ITF category and are synonymous. They are designed to characterize 1) the overall RCS-system behaviors, stand-alone and under the influence of coupling; 2) the overall PCV-system behaviors under the influence of coupling; 3) the overall coupled system behavior and the related phenomena and processes; 4) the interactions of two or more components; 5) local phenomena. Please refer to the Notes below.

Notes:

In the current reactor design it is possible to study the PCV physical behavior separately from the RCS-physical behavior, NEA/CSNI, 1996b. The RCS is the source of water/steam and hydrogen for the PCV and can be considered as a boundary condition for the PCV analyses, NEA/CSNI, 1999. In the new advanced passive reactor designs it is not possible to consider the RCS as a boundary condition for the PCV but it is necessary to consider the physical behavior of the PCV coupled with the RCS physical behavior, NEA/CSNI, 1996b. It is necessary to characterize the RCS/PCV coupled behavior during the transient evolution. This is due to the strong coupling effects and feedback between the RCS and PCV. The passive mitigation strategy is based on natural circulation loop covered both components to remove the decay heat.

SCALE-DOWN
This is to design a test facility with smaller size and/or lower pressure/temperature conditions compared with those conditions of reference reactor by using an appropriate scaling method. Utilization of simulant has similar meaning.

SCALE EFFECT
Consequences due to scaling distortion in test conditions of and/or test results from scaled test facility (scaled model)

SCALE-UP
This is to extrapolate phenomena, local and/or system-wide, observed in a scaled-down test facility to prototype phenomena that may appear in the reference facility.

SCALING
'Scaling', 'scaling issue' and 'addressing the scaling issue' are key terms adopted in nuclear reactor design and safety. They indicate the actions, the methods and the approaches aimed at connecting the parameter values related to experiments or to computational tool calculations with Nuclear Power Plant (NPP) conditions; the concerned parameter values are applicable and qualified under the reduced-scale conditions; the reduced-scale conditions imply values of geometry, pressure, or power, or combinations, smaller than the values characterizing the NPP conditions.

Scaling is the process of converting any plant parameters at reactor conditions to those either in experiments or in numerical code results in order to reproduce the dominant prototype phenomena in the model. Scaling issue indicates the difficulty and complexity of the process and the variety of connected aspects. Addressing the scaling issue refers to a process of demonstrating the applicability of those actions performed in scaling.

SCALING ANALYSIS
The analysis describing the scaling objectives, methodology and distortions of the scaling performed on a scaled model according to the design of the prototype. Typically the analysis describes the assumptions and justifications of the methodology based on theoretical derivations.

SCALING APPROACH
A technical (and economical) strategy to provide a design of scaled test facility (scaled model) that may give the best simulation of reference reactor under certain initial and boundary conditions (within given budget). All the relevant and necessary processes and phenomena should be identified before the design work based on engineering judgement, code evaluation and experiences. Scaling technique is used to attain the scaling approach. Meaning of Scaling Concept, Scaling Strategy and Scaling Principle is almost equal to that of Scaling Approach.

SCALING CAPABILITY
Capability to connect a parameter value measured in an experiment or calculated by a code and the equivalent reference system (typically the NPP) parameter value.

SCALING CRITERIA
See Scaling Method.

SCALING DISTORTION
Any deviation between a reference system (NPP) parameter value and the same value calculated by a model or measured in an experiment constitutes a scaling distortion. A specific problem arises when the reference system parameter value is unknown and the related distortion needs to be quantified.

SCALING EVALUATION
Analyses to identify origin of and to quantitatively evaluate influences due to scaling distortions that arise in tests performed by using a scaled test facility (scaled model)

SCALING FACTOR
A ratio of system variable or parameter in scaled system to that in prototypical system. This may include volumetric (scaling) ratio Kv etc. -- (This is different from scale factor in cosmology and such other domains of science than fluid mechanics)

SCALING ISSUE
Scaling issue indicates the difficulty and complexity of the scaling process and the variety of connected aspects. Please see the “Scaling” definition.”

SCALING LAW
Mathematical relationship, ratio and/or rate to best describe scaling factor of thermal-hydraulic phenomena simulated in a test facility under certain initial and boundary conditions designed to best simulate prototype phenomena in reference reactor. A set of scaling laws is used to form a scaling method on which the test facility is designed. Scaling laws designate a set of conditions that assure complete similarity between the scaled and prototype systems as long as linear relationship is realized within the range of application. Froude number (Fr), Reynolds number (Re), Weber number (We) are included in scaling laws. New scaling laws associated to local and/or system-wide phenomena are sometimes defined according to objectives to scale specific prototype phenomena.

SCALING LIMIT
A condition (status) of an attained range of test conditions that do not fully cover the range of prototype conditions. Scaling limit is a part of the scaling distortion.

SCALING METHOD
A method to design a test facility following a given scaling approach (concept, strategy, principle) to best simulate aimed thermal-hydraulic response of local phenomena and/or system response of reference reactor, to specify test initial and boundary conditions, to understand the observed test results and to make the test results applicable to the reference reactor with evaluations of uncertainty in the applied (extrapolated or interpolated) results. It is composed of several scaling laws. Test facility is, in most cases, scaled-down by using a scaling method. Meaning of Scaling Criteria and Similarity Criteria is similar to that of Scaling Method, while the former two are somehow specific to a certain parameter. This may include H2TS etc.

SCALING PARAMETER
An element used for scaling laws and scaling factors to describe each of processes or phenomena in a system of interest

SCALING SIGNIFICANT
A term associated with SETF and ITF design. A facility is considered scaling significant when the best connection can be established between the facility layout and the measured test scenarios and the layout of the prototype or the expected test scenario in the prototype, respectively.

SCALING TECHNIQUE
A technique to pursue and achieve a planned scaling approach by using some of scaling methods

SIMILAR TEST
Experiments performed in differently scaled models, i.e. models designed, constructed and operated following different sets of scaling factors. In the case of ST one reference experiment should exist and distortions in BIC values are expected with consequences quantified.

SIMILARITY CRITERIA
See Scaling Method.

SIMILARITY PARAMETER
See Scaling Parameter.

SPECIAL COUNTERPART TEST
See Daughter Test.

ABBREVIATIONS & ACRONYMS
(*USED IN THE MAIN TEXT AND IN A-1, A-2 AND A-4*)

ABWR	Advanced BWR (BWR is already defined)
ACC	Accumulators
ACRS	USNRC Advisory Committee on Reactor Safety
ADS	Automatic Depressurization System
AEC	*see* US AEC
AFW	Auxiliary Feed-water
ALARA	As Low As Reasonably Achievable
AM	Accident Management
AMP	Accident Management Procedure
ANL	Argonne National Laboratories
ANS	American Society of Nuclear Engineering
AOC	Agents of Change
AP-1000	Advanced PWR
AREVA	Designer of NPP (Company)
ARL	Applied Research Laboratory
ARN	Autoridad Regulatoria Nuclear (Argentina)
ASME	American Society of Mechanical Engineering
ATHLET	Analysis of THermal-hydraulics of LEaks and Transients (computer code)
ATLAS	2-active loop ITF available in Korea to simulate PWR (RHFP)
BC	Boundary Conditions (see also BIC)
BCL	SETF available in US for CCFL studies
BE	Best Estimate
BEAU	Best Estimate Analysis and Uncertainty
BEMUSE	Best Estimate Methods plus Uncertainty and Sensitivity Evaluation
BEPU	Best Estimate Plus Uncertainty
BETHSY	3-active-Loop ITF available in France to simulate PWR (FHFP)
BDBA	Beyond Design Basis Accident
BFC	Containment facility available in Germany
BFMC	Battelle-Frankfurt Model Containment (experimental facility)
BIC	Boundary and Initial Conditions
BNL	Brookhaven National Laboratory
BPG	Best Practice Guidelines
BWR	Boiling Water Reactor
B&W	Babcock and Wilcox
CANDU	Canadian Deuterium Uranium
CAER	Center for Advanced Engineering Research
CAPS	CSNI Activity Proposal Sheet
CATHARE	Code for Analysis of Thermal hydraulics during an Accident of Reactor and safety Evaluation
CCF	Counter-current flow

CCFL	Counter-Current Flow Limitation
CCTF	Cylindrical Core Test Facility: Large scale SETF for 2D-3-D Program: available in Japan
CCVM	(CSNI) Computer Code Validation Matrix
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
CET	Core Exit Thermocouple
CFD	Computational Fluid Dynamics
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CIAU	Code with capability of Internal Assessment of Uncertainty
CL	Cold Leg
CLI	Cold Leg Injection
CMT	Core Make-Up Tank
CNEN	Comissão Nacional de Energia Nuclear (Brazil)
COSI	SETF available in France for DCC studies
CPU	Central Processing Unit
CPV	Cooling Pool Vessel
CREARE	SETF available in US for CCFL studies
CRDM	Control Rod Drive Mechanisms
CRGT	Control Rod Guide Tube
CSAU	Code Scaling, Applicability and Uncertainty
CSNI	Committee on the Safety of Nuclear Installations
CT	Counterpart Test(s)
DBA	Design Basis Accident
DC	Downcomer
DCC	Direct Contact Condensation
DCH	Direct Containment Heating
DiD	Defense in Depth
DNB	Departure from Nucleate Boiling
DNS	Direct Numerical Simulation
DSA	Deterministic Safety Analysis
DSS	Dynamical System Scaling
DVI	Direct Vessel Injection
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EDO	EDO "Gidropress", Russian organization based in Podolsk
EM	Evaluation Model
EMDAP	Evaluation Model Development and Assessment Procedure
EMSI	European Multiphase System Institute
ENEA	Italian agency for new technologies, energy and sustainable economic development (Agenzia Nazionale per le nuove tecnologie, l'Energia e lo sviluppo economico sostenibile)
EPR	European Pressurized Reactor
EPRI	Electrical Power Research Institute
ERSEC	SETF available in France for reflood studies
ETN	Utility for NPP in Brazil
ETPFGM	European Two-Phase Flow Group Meeting
FEBA	Flooding Experiments with Blocked Arrays
FHFP	Full Height Full Pressure
FOM	Figure of Merit
FRC	Fractional Rate of Change

FSA	Fractional Scaling Analysis
FSAR	Final Safety Analysis Report
FW	Feed water
GE	General Electric - Reactor Designer
GEST	Generator Separator Test (experimental facility)
GRS	Gesellschaft Fur Reaktosicherheit
HDR	Containment full scale facility available in Germany
HEM	Homogeneous Equilibrium Model
HPIS	High Pressure Injection System (see also HPSI)
HTC	Heat Transfer Coefficient
H-F	Henry-Fauske
HL	Hot Leg
HPC	High Pressure Containment
HPSI	High Pressure Safety Injection (see also HPIS)
HSSE	Hot Side Straight Effect
H2TS	Hierarchic 2-Tiered Scaling
HX	Heat exchanger
IAC	Interim Acceptance criteria of US AEC
IAEA	International Atomic Energy Agency
IC	Initial Conditions (see also BIC)
ICS	Isolation Condenser System
ICSP	International Collaborative Standard Problems (from IAEA)
IET	*see* ITF
IETF	Integral-effect test facility
IIST	RHRP facility available in Taiwan
IRIS	Innovative Reactor designed by Westinghouse
IRWST	In-containment Reactor Water Storage Tank
ISP	International Standard Problem (from NEA/CSNI)
ITF	Integral Test Facility
J. (or J)	Journal
JAEA	Japan Atomic Energy Agency
KAERI	Korean Atomic Energy Research Institute
KARLSTEIN	SETF available in Germany for centrifugal pump studies
KERENA	1250 MWe Boiling Water Reactor (Areva)
LBLOCA	Large Break LOCA
L/D	length-to-diameter ratio
LES	Large Eddy Simulation
LLC	Limited liability company
LOBI	2-active-Loop ITF available in Italy to simulate PWR
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test: 1-active-Loop ITF available in US to simulate PWR
LP	Lower plenum
LSTF	Large-Scale Test Facility: 2-active-Loop ITF in Japan to simulate PWR
LUT	Look-Up Tables (for CHF)
MARVIKEN	Large-Scale SETF available in Sweden for blowdown studies
MELCOR	Accident simulation code
MHI	Mitsubishi Heavy Industry
MIST	Multi-loop Integral System Test: 2-active-Loop ITF available in US to simulate PWR with OTSG
M.M.	Man-Month
MSLB	Main Steam Line Break

MW	Mega Watt
NA-SA	Utility for NPP in Argentina
NC	Natural Circulation
NCFM	Natural Circulation Flow Map
NEA	Nuclear Energy Agency
NED	Nuclear Engineering and Design
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission (US)
NRS	Nuclear Reactor Safety
NST	Nuclear reactor Safety Technology
NVG	Net Vapor Generation
ODE	Ordinary Derivative Equations
OECD	Organisation for Economic Co-operation and Development
ONB	Onset of Nucleate Boiling
OTSG	Once-Through Steam Generator
PBL	Pressure Balance Line
PCCS	Passive Containment Cooling System
PCT	Peak Cladding Temperature
PCV	Primary Containment Vessel
PDE	Partial Derivative Equation
PERICLES	SETF available in France for reflood studies
PERSEO	Thermal-hydraulic test facility (SIET, Piacenza, Italy)
PHWR	Pressurized Heavy Water Reactor
PIPER-ONE	ITF available in Italy for SBLOCA in BWR
PIRT	Phenomena Identification and Ranking Table
PKL	4-active-Loop ITF available in Germany to simulate PWR
PMK	Paks Model Circuit (in Hungarian), experimental facility
PRG	Programme Review Group
PRHR	Passive Residual Heat Removal
PRZ	Pressurizer
PSA	Probabilistic Safety Assessment
PSB	4-active-Loop ITF available in Russia to simulate VVER
PSI	Paul Scherrer Institute
PSP	Pressure Suppression Pool
PSS	Passive Safety System
PTS	Pressurized Thermal Shock
PUMA	Facility to simulate SBWR available at Purdue University (RHRP)
PV	Pressure Vessel, see also RPV
PVST	Power-to-Volume-Scaling Tool
PWG-2	NEA/CSNI principal working group for accident analysis
PWR	Pressurized-Water Reactor
RANS	Reynolds Averaged Navier Stokes
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RNG	Physical model in CFD code
RG	Regulatory Guide
RHFP	Reduced-Height, Full-Pressure
RHRP	Reduced-Height, Reduced-Pressure
ROSA	Rig Of Safety Assessment
RPV	Reactor Pressure Vessel
RSA	Relative Scaling Analysis

RST	Rod Surface Temperature, or physical model in CFD code
RD	Canadian thermal-hydraulic experimental facility
R&D	Research and Development
REBECA	Mist flow between sub-compartments of a containment experimental programme (France)
REBEKA	REactor typical Bundle Experiment KArlsruhe
ROCOM	ROssendorf COolant Mixing experimental facility
SA	Severe Accident
SARNET	Severe-Accident Research NETwork
SBLOCA	Small-Break LOCA
SBO	Station Blackout
SBWR	Simplified BWR
SCTF	Slab Core Test Facility, Large scale SETF for 2D-3-D Program: available in Japan
SDA	Standard Design Approval
SETF	Separate Effect Test Facility
SEMISCALE	Two-active-Loop ITF available in US (Idaho) to simulate PWR (FHFP)
SG	Steam Generator
SI	Safety Injection
SMR	Small Modular Reactor
SOAR	State of Art Report
SPOT	Passive heat removal system (in Russian)
SPES	Three-active-Loop ITF available in Italy to simulate PWR (FHFP)
SQE	Software Quality Engineering
SRS	Safety Report Series (IAEA)
SSG	Specialists Scaling Group, or (IAEA) Specific Safety Guide
S-SOAR	Scaling SOAR
SST	Physical model in CFD code
ST	Similar Test(s)
SWAT	SMART ECC Water Asymmetric Two-phase choking test facility
SYS	System
TECC	Thermal Hydraulics of Emergency Core Cooling
TF	Test facility
TH	Thermal-Hydraulics
THETIS	Thermal-Hydraulics Experiments on a model PWR fuel assembly
TMI	Three Mile Island
TOSQAN	TOnus Qualification ANalytique, French experimental apparatus
TPCF	Two-Phase Critical Flow
TPPD	Two-Phase Pressure Drops
TPTF	Two-Phase Flow Test Facility
TRAC	Transient analysis computer code (US NRC)
UH	Upper Head
UM	Uncertainty Methods, or 2-active-Loop ITF available in US to simulate PWR
UMAE	Uncertainty Methodology based on Accuracy Extrapolation
UMS	Uncertainty Method Study
UNIPI	University of Pisa
UPC	Polytechnic University of Catalunia
UPTF	Large scale SETF: full scale RPV and simplified RCS, available in Germany
UQ	Uncertainty Quantification
USAEC	United States Atomic Energy Commission
USNRC	United States Nuclear Regulatory Commission
UVUT	Unequal Velocities, Unequal Temperatures
VANAM	German experiments performed in the Battelle Model Containment

VVER	Water-Cooled, Water-Moderated Energy Reactor or Vodo-Vodyanoi Energetichesky
V & V	Verification and Validation
WGAMA	NEA/CSNI Working Group on Analysis and Management of Accidents
2D/3-D	Experimental research programme based on UPTF, SCTF and CCTF facilities

APPENDICES

A-1 THE NEA/CSNI/WGAMA CAPS FOR THE PRESENT ACTIVITY WGAMA (2013)2

Project/Activity Title	State-of-the-Art Report on Scaling of Thermal-hydraulic Systems Using Water as Working Fluid during Design Basis Accidents
Objective	The objective is to summarize the technical knowledge suitable for addressing the scaling issue in the licensing of existing Nuclear Power Plants (NPP) and NPPs currently under construction or in advanced design stage.
Scope	A Special Scaling Group (SSG) was constituted as a follow-up of the 15th WGAMA meeting and a preparatory (activity-launching) meeting was held in June 2013. The scope for the activity formulated includes the following: 1. Restrict scaling to water as the working fluid; 2. Exclude scaling issues relating to severe accidents involving loss of core integrity; 3. Activity limited to NPPs in operation and under construction or in advanced design stage; and 4. Focus on thermal-hydraulics only and not addressing fluid structure interaction namely flow induced vibrations.
Justification	The reasons for the activity include: a) ‘Addressing the scaling issue’ is a key activity in licensing. The ‘issue’ became more important when the Best Estimate Plus Uncertainty (BEPU) approach was pursued in meeting the requirements of Chapter 15 of the Final Safety Analysis Report (FSAR) of USNRC for individual NPP and in other connected chapters. b) Scaling is the process of demonstrating how and to what extent the numerical simulation tool validated on one or several reduced scale experiments (or at different values of some flow parameters such as pressure and fluid properties) can be applied with sufficient confidence to the real process such as in a NPP. The ‘issue’ comes from the fact that phenomena or system performance expected in NPPs cannot be replicated under the same conditions of power, volume and pressure which characterize the NPP. A computational tool can be validated only against ‘scaled’ data. This creates concerns which are synthesized by the words ‘scaling issue’. c) Views on ‘scaling’ differ and therefore there is no consensus on the application among designers and scientists. d) Internationally agreed documents about scaling views have not been issued.
Expected results and deliverables	The deliverable will be a State-of-Art-Report summarizing the technical knowledge suitable for addressing the scaling issue in the licensing of existing Nuclear Power Plants (NPP) and NPPs currently under construction or in advanced design stage.
Users	The users will be regulators, technical safety organizations, designers and operators involved in the application of best estimate computer codes. Code

	developers could also benefit from the report.
Relation to other projects	This project is fully integrated within the streamline of WGAMA activities dealing with thermal-hydraulics. The activity is also related to Verification and Validation (V & V) of computer codes. The International Standard Problems (ISP) activity and the SOAR on Thermal-hydraulics of Emergency Core Cooling System (SOAR on TECC) are directly connected to the proposed activity. There is direct connection with the international Counterpart Test (CT) activities: part of that has been performed in the framework of NEA/CSNI. The OECD 'PKL' and 'ROSA' Projects and the recently launched 'ATLAS' Project has direct interest in the present activity. Scaling is also related to BEPU activities, including joint activities between NEA and IAEA.
Safety significance/ priority (see priority criteria in Section IV of the CSNI Operating Plan)	BEPU approach in licensing requires addressing and proposing a solution to the scaling issue. Validation of codes used in NPP safety analysis implies achieving the same goal. The activity corresponds to the following CSNI criteria: - Criterion 2: Better accomplished by international group - Criterion 3: Likely to bring results in a reasonable time frame - Criterion 4: Maintain and preserve strategic safety competence
Technical Goals covered	3-D) To assess advanced methods and tools used for event/accident analysis; [….]; to quantify corresponding uncertainties 4c) To review current analytical tools as well as risk assessment approaches regarding their applicability to safety assessments of new designs and further develop and validate them where needed 5c) To review the adequacy of analytical tools [….] and validate new analytical approaches when called for by specific features of new designs.
Knowledge management and transfer covered	The task results will be documented in a State-of-the-Art Report that may be used for knowledge transfer.
Milestones (deliverables vs. time)	Three meetings will be organized in two years. The State-of-the-Art Report (SOAR) will be issued at the end of the activity. Time schedule proposed is: • December 2013 or January 2014: **First Meeting of SSG**: o Agreement on the scope details and on the list of content of the SOAR • June 2014: **Second Meeting of SSG**: o Draft content of each Chapter. • September 2014: Presentation to WGAMA of current progress of activities • January 2015: **Third Meeting of SSG** with all agreed contributions approved. • September 2015: Presentation to WGAMA of the draft final Report • End of 2015: Deliverable submission to PRG and CSNI
Lead organization(s) and coordination	University of Pisa – Italy
Participants (individuals and organizations)	The present 'configuration' of SSG includes the following participants (Organization and individuals): 1. AREVA (Germany) : H. Schmidt (attending in place of K. Umminger) 2. CEA (France): D. Bestion 3. GRS (Germany): H. Glaeser

	4. JAEA (Japan): H. Nakamura 5. KAERI (Korea): H-S. Park 6. PSI (Switzerland): O. Zerkak 7. UPC (Spain): F. Reventos 8. NRC (USA): P. Lien 9. UNIPI (Italy): F. D'Auria, M. Lanfredini, N. Aksan
Resources	Each SSG member (or replacement) should be available for 2 M-M equivalent full time work during the 2 years period. Desirable but not essential: Financial support may be needed for the work of N. Aksan (2 M. M. and 3 meetings)
Requested action from PRG/CSNI	Endorsement
PRG Recommendation	Endorsed
CSNI Disposition	Approved

A-2 AN OUTLINE OF THE HISTORY OF SYS TH AS A BACKGROUND FOR SCALING

Taken from D'Auria, 2012

The history of nuclear thermal-hydraulics shall start with the E. Fermi pile in 1942 (Dec. 2nd): several tons of graphite were assembled in the pile having a cross section > 20 m^2 in order to ensure suitable thermal capacity for dissipating the thermal power possibly produced by the chain fission reaction (other than to minimize neutron leakages). Afterwards, the design, construction, and operation of energy systems in the range from a few KW to thousands of MW were possible with a parallel and consistent development of the thermal-hydraulics discipline.

Before 1960

Accidents and related scenarios in nuclear power plants were considered to demonstrate the safety of NPP in the 1950s when computers did not exist. Experiments, pioneering thermal–hydraulics models, and engineering evaluations were the basis of reactor safety analyses at the time. Nuclear thermal-hydraulics and reactor physics (or neutron kinetics), as well as nuclear fuel materials, were the subjects of integrated studies.

1960-1970

Systematic thermal–hydraulic studies and experiments were conducted in the 1960s, noticeably concerning individual 'physical' phenomena like two-phase critical flow, critical heat flux, depressurization and blow-down. Thermal-hydraulics became a 'self-standing' discipline. Several small scale fundamental programs were launched and completed. New findings from those research projects were considered in reactor safety and licensing documents.

1970-1980

Massive use of computers for nuclear reactor safety started in the 1970s. The accident analysis could also benefit from primitive (SYS TH) numerical codes and from measurements taken in integral-system experiments. The nuclear regulatory point-of-view was established with the publication of the 'Interim Acceptance Criteria for ECCS' (1971), USAEC, 1971. This triggered a wide variety of research aimed at the evaluation of safety margins and focusing on the estimation of the maximum temperature on the surface of fuel rods following large break loss of coolant accidents (LB-LOCA). Appendix K to paragraph 10 CFR-50.46 (Code of Federal Regulation) followed in 1974. The publication of the 'Interim Acceptance Criteria for ECCS' shall be taken as the starting date for SYS TH: competences were requested to comply with those criteria. The technological community and the industry reacted to the request by regulators: comprehensive research projects were started in the experimental area as well as in the area of code development. Large experimental facilities were designed and operated and the SYS TH codes were made available for transient analyses in NST.

Large scale experimental ITF implied international cooperation projects and 'relevant' measured data were gathered to understand transient system thermal-hydraulic performance. TPCF (Two-Phase Critical Flow) and CHF/DNB (Critical Heat Flux / Departure from Nucleate Boiling) can be identified as the key thermal-hydraulic phenomena of interest during the decade associated with the LBLOCA (Large Break Loss of Coolant Accident) event.

'Conservatism' is the keyword which characterizes the application of Appendix K (to 10 CFR 50.46) in licensing analyses. During the same decade WASH-1400 or the "Rasmussen Report" was issued, putting the basis for the application of PSA in NST; significant results from the execution of probabilistic analyses were produced, USNRC, 1975. At the end of the decade, in 1979, the Three Mile Island Unit 2 accident happened. In the area of SYS TH, this shifted the attention from LBLOCA to SBLOCA phenomena.

1980-1990

Within the framework of SYS TH code use, V & V (Verification and Validation) was soon recognized, e.g. D'Auria & Galassi, 1998, as a mandatory process to be completed before application of those computational tools to safety and licensing. In this context, the basis was set for addressing the scaling issue, e.g. D'Auria & Galassi, 2010.

The reference SYS TH codes are APROS**, ATHLET*, CATHARE*, KORSAR*, MARS**, RELAP*, SPACE**, TRAC*, TRACE**, where: * = precursor code and ** = lately developed code.

International activities were conducted at CSNI (Committee on the Safety of Nuclear Installations of OECD/ NEA, Organization for Economic Cooperation and Development/Nuclear Energy Agency) proposing viable ways for V&V, NEA/CSNI, 1987, NEA/CSNI, 1993, and NEA/CSNI, 1996. The importance of user effect upon the predictions was recognized, Aksan et al., 1993, NEA/CSNI, 1998, and D'Auria, 1998, as well as the role of the input deck (or nodalization) and of the related qualification, e.g. Bonuccelli et al., 1993. The contribution to the understanding of important NST phenomena from ITF experimental programs, conducted or initiated during this decade shall be realized. Key acronyms for the ITF or large scale SETF, within BWR, PWR and CANDU technologies (related research programs may have developed in decades different from the current one in the cases identified by an asterisk: however, for the sake of synthesis all major research programs in SYS TH are listed hereafter in alphabetic order), are: APEX*, ATLAS*, BETHSY, CCTF, FIST, FIX-II, GIRAFFE, HDR, ISB, LOBI, LOFT, LSTF, MARVIKEN, MIST, PACTEL, PANDA, PIPER-ONE, PKL, PMK, PSB*, PUMA, RD-14M, ROSA, SCTF, SEMISCALE, SPES, UM, THTF, and UPTF, where: * = lately constructed facility. In this framework, the 2D/3-D international cooperation program, USNRC, 1993, (involving the already mentioned UPTF, SCTF and CCTF), provided key information to address the scaling issue, i.e. connecting the measured data with expected NPP conditions, from the experimental viewpoint. Enormous benefits were gained in the area of demonstrating SYS TH code capabilities.

CCFL (Countercurrent Flow Limitation) and NC (Natural Circulation) can be identified as the key thermal-hydraulic phenomena of interest during the decade associated with the SBLOCA (Small Break Loss of Coolant Accident) event, together with reflood. The thermal-hydraulics of ECCS (Emergency Core Cooling Systems) shall be mentioned in this context.

The words SYS TH were proposed inside the scientific community. Later on, those words became of common use. The need for uncertainty methods suitable for predicting unavoidable errors to be added to the results of calculations performed by system thermal–hydraulic codes became clear at the beginning of the 1990s (or even at the end of the 1980s). Working approaches were proposed; noticeably, the pioneering effort by USNRC shall be mentioned, USNRC, 1989, which lead to the formulation of CSAU (Code Scaling and Applicability Uncertainty). The PIRT process was proposed.

In the middle of the decade, in 1986, the Chernobyl Unit 4 accident happened. In the area of SYS TH, this moved increased attention toward passive systems and the processes for the design of AP-600 and SBWR had a strong impulse.

1990-2000

Addressing the uncertainty in SYS TH as a follow-up of V & V was the center of attention in this period. Following and considering the CSAU, the Wilks formulation and the UMAE (Uncertainty Methodology based on Accuracy Extrapolation) were proposed, Hofer, 1990, and D'Auria et al., 1995 (already available to scientific community in 1993). The UMS (Uncertainty Method Study) project was launched by the

CSNI in 1993 and completed in 1998, NEA/CSNI, 1998a: the fundamental features of the uncertainty methods were described in detail and a suitable demonstration was achieved in relation to their robustness and qualification level. The USNRC issued the Regulatory Guide (RG) 1.157, USNRC, 1989a: the application of system thermal–hydraulic codes was envisaged, even though recommending the use of selected conservative models. Those models are concerned with phenomenological areas where the knowledge was not considered satisfactory. Requirements in the RG 1.157 did allow a few attempts at practical applications. However, Appendix K to 10 CFR 50.46 continued to be used during the decade for licensing purposes. The acronym BEPU (Best Estimate Plus Uncertainty) started to circulate. A break-through workshop for planning the future in SYS TH was held in Annapolis (1996) under the combined effort by NEA and US NRC. The development of a new SYS TH code was launched (current name is TRACE), following identification of inadequacies in existing codes at the time. Notably, the key words 'Internal Assessment of Uncertainty' were proposed during the workshop. At the end of the decade, the CIAU method (Code with capability of Internal Assessment of Uncertainty) was developed, D'Auria & Giannotti, 2000, and ready for practical applications. CIAU used UMAE (mentioned before) as the 'engine' and the qualification tool for the process of code application. The development of CFD (Computation Fluid Dynamic) technology, mostly connected with single-phase flows, had a strong impulse, and made possible with the increase in computational power, Yadigaroglu et al., 2003. The SYS TH coupling with three-dimensional neutron physics was also made possible by the availability of more powerful computers and numerical techniques, NEA, 2004.

2000-2010

Application of BEPU approaches in licensing processes, implying the exploitation of the capabilities of SYS TH codes and of UM, definitely started in the 2000s. The following key events, not an exhaustive list, not in the order of importance, not in the order of time, see also Petruzzi et al., 2005, give an idea of the technology developments in the area: a) The AREVA (NPP designer) on the behalf of the ETN (Brazilian utility owner for the nuclear plant in Angra) proposed a BEPU methodology to analyse the LBLOCA for the licensing of Angra-2 NPP in Brazil, KWU-Siemens, 1997. The submission was analysed by the regulatory authority of Brazil which also requested the application of different uncertainty methods by assessors, independent from AREVA. b) USNRC issued the RG 1.203, USNRC, 2005, which provided clarification of the regulatory expectation for transient and accident analysis including the application of BEPU approaches. c) CSNI launched and completed the six-year project BEMUSE. The aim was to demonstrate the maturity of uncertainty methods and approaches with main concern to LBLOCA applications. The objective was achieved, but differences in the results by participants (mainly in predicting reflood time) caused the need for a careful interpretation of related findings. The difficulty in harmonizing, from the side of applicants of uncertainty methods, the choice of input uncertainty parameters and the related ranges of variations was an outcome from the project. d) Three important BEPU-concerned documents were issued by IAEA, two Safety Report Series, SRS 23 and SRS 52, IAEA, 2002, and IAEA, 2008, and one Specific Safety Guide, IAEA 2010. The SRS 52 deals with the description of workable uncertainty approaches and methods. The SSG-2, dealing with Deterministic Safety Analysis (DSA) in general, proposes the BEPU approach in licensing as consistent with the technological state of the art in the area of accident analysis. e) Best estimate (BE) conferences, BE-2000 and BE-2004, ANS, 2000 and ANS, 2004, were held under the auspices of the American Nuclear Society (ANS). This series of conferences was actually continued by V & V Workshops in the US in Idaho Falls (Idaho) in 2008, Myrtle Beach (North Carolina) in 2010, and Las Vegas (Nevada) in 2012, with the cooperation of the nuclear sector of the ASME. f) The BEAU (application of the Best Estimate Analysis and Uncertainty) method was proposed by Canadian experts, Abdul-Razzak, 2009. g) A variety of BEPU (it shall be clear that the BEPU acronym is not always adopted) applications all over the world during the concerned decade, mostly within the license renewal framework, are summarized by Glaeser, 2008.

The first decade of the current millennium is characterized by the application in NST of the expertise in thermal-hydraulics: the BEPU approach constitutes the key word in this connection. The 2010s decade started with the submission of Chapter 15 of the FSAR (Final Safety Analysis Report) of the Atucha-II

NPP to the regulatory authority in Argentina, by the NA-SA utility. In this case, the entire chapter of the FSAR is based on the BEPU and the approach itself was submitted in advance and endorsed by the regulatory body, UNIPI-GRNSPG, 2008. Key words for the TH spot historic outline can be found in Table A2-1. In the first row, the key ‘actors’ and the ‘stakeholders’ for the nuclear thermal-hydraulics discipline are listed.

<table>
<tr><th colspan="2">ACTORS & STAKE-HOLDERS* =></th><th>Authors of Textbooks, US NRC, International & National Institutions, Industry, NURETH (Conferences), Journals</th></tr>
<tr><th>PERIOD OR EVENT</th><th>KEY WORDS/PHENOMENA</th><th>KEY DOCUMENT</th></tr>
<tr><td>Fermi Fission Reaction (1942)</td><td>Thermal Capacity (of graphite)</td><td></td></tr>
<tr><td>Up to 1960</td><td>Heat Transfer & Pressure Drops</td><td>E.g.: Dittus-Boelter eq. for HTC, Multiplier Approach for TPPD</td></tr>
<tr><td>1960-1970</td><td>TH Fundamentals; TPCF; Blow-down; CHF/DNB</td><td>E.g.: Moody and H-F models for TPCF, LUT for CHF</td></tr>
<tr><td>1970-1980</td><td>LBLOCA – Conservatism; TPCF; CHF/DNB; Code Design</td><td>USNRC IAC for ECCS, App. K to 10 CFR 50.46</td></tr>
<tr><td>1980-1990</td><td>SBLOCA – BE; V & V & Scaling; 2D/3-D; CCFL; NC; Code Validation</td><td>CSAU, USNRC Compendium, CSNI SOAR on TECC, CCVM-ITF</td></tr>
<tr><td>1990-2000</td><td>AM; CFD; UM; Code Validation & Application</td><td>CCVM-SETF, UMS**, USNRC RG 1.157, UMAE, GRS-method</td></tr>
<tr><td>2000-2010</td><td rowspan="2">Licensing; BEPU (Code Application) & Scaling; Passive System Thermal-hydraulics</td><td rowspan="2">USNRC RG 1.203, IAEA SRS 23 and 52, IAEA SSG-2, BEMUSE**, NURESIM**, NURISP**, CASL**</td></tr>
<tr><td>2012</td></tr>
<tr><td>After 2012</td><td>Consolidation in the above areas***</td><td>NURESAFE**, CASL, PREMIUM**</td></tr>
</table>

See Introduction, **Acronyms for and International Project, ***See Chapter 3

Table A2-1 - Actors and Stake-holders in SYS TH: a historical excursus and key documents (taken from D’Auria, 2012).

A-3 LIST OF SELECTED SETF AND ITF

The Appendix A-3 provides tables of test facilities of SETF and ITF used to obtain data to search for possible phenomena during reactor accident, to understand thermal-hydraulic response of LWR system, to understand local phenomena in detail, to develop models and correlations, to validate computer codes, etc.

The tables in A-3 provide unique set of information on the experimental facilities necessary for a better understanding of chapter 3.2. These tables are listed in Table A3-0. Examples of items directly or indirectly considered in the tables as relevant to scaling are: name, type, objective(s), main phenomena investigated, reference reactor, facility geometry, scaling method, test fluid, parameter range, scaling ratio, organization and country, of each test facility, with major references. Scaling distortion may be estimated from the scaling ratio (power, volume, pressure and geometry such as height, diameter etc.) to the reference reactor and possible change in the observed phenomena from the "expected" phenomena in reactor.

Table A3-0 – List of tables in current Appendix.

Table No	Key Content	Notes
A3-1	**ITF constructed and operated for the simulation of RCS including advanced designs**	Country where the facility is installed and Owner Institution are identified.
A3-2	**ITF simulating PWR, main characteristics**	Including those considered in NEA/CSNI, 1989, see also Karwat, 1985. Information from recent research programs has been added.
A3-2A	**ITF simulating PWR, pump data**	
A3-2B	**ITF simulating PWR, core data**	
A3-2C	**ITF simulating PWR, SG data**	
A3-3	**ITF simulating BWR, main characteristics**	
A3-3A	**ITF simulating BWR, core data**	
A3-4	**ITF and SETF simulating VVER, main characteristics**	Considered in NEA/CSNI,2001
A3-4A	**ITF and SETF simulating VVER, pump data**	
A3-4B	**ITF and SETF simulating VVER, core data**	
A3-4C	**ITF and SETF simulating VVER, SG data**	
A3-5	**ITF simulating advanced PWR, main characteristics**	
A3-5A	**ITF simulating advanced PWR, pump data**	
A3-5B	**ITF simulating advanced PWR, core data**	
A3-5C	**ITF simulating advanced PWR, SG data**	
A3-6	**Facilities constructed and operated for the simulation of CONTAINMENT including advanced designs**	Country where the facility is installed and Owner Institution are identified.
A3-7	**CONTAINMENT facilities, main characteristics**	Including those considered in NEA/CSNI, 1989a, and NEA/CSNI 1999
A3-8	**Pressure Suppression CONTAINMENT systems, comparison of selected data**	Considered in NEA/CSNI, 1986
A3-9	**Pressure Suppression CONTAINMENT facilities, comparison of selected geometrical data**	
A3-10	**Pressure Suppression CONTAINMENT facilities, key scaling factors**	

Table A3-1 - ITF constructed and operated for the simulation of RCS including advanced designs.

Country	ITF	Organization	Country	ITF	Organization
Canada	RD-14M+	CNL Whiteshell Laboratories	**Netherlands**	DESIRE	Delft University of Technology
Finland	PACTEL*	VTT Energy		CIRCUS	
	PWR PACTEL		**Russia**	KMS	NITI
	REWET-III			PSB-VVER	EREC
France	BETHSY*	CEA		ISB-VVER	
	CLOTAIRE			PM-5	IPPE
Germany	PKL*	AREVA		SB	EDO-Gidropress
Hungary	PMK-2	KFKI-AEKI		BD	
Italy	LOBI*	EC- EURATOM	**Sweden**	FIX-II*	Studsvik
			Switzerland	PANDA	PSI
	SPES*	SIET	**US**	LOFT*	INEL
	SPES-2			SEMISCALE*	
	SPES-3 (under costruction)			Gerda	B & W
	PIPER-ONE*	Univ. of Pisa		OTIS*	
Japan	ROSA-III*	JAERI (JAEA)		MIST	
	LSTF-(ROSA-IV)*			UMCP	Univ. of Maryland
	ROSA-AP600 (LSTF)			TLTA*	GE
	CCTF			FIST*	
	TBL*	Hitachi		GIST	
	GIRAFFE	Toshiba		APEX	Oregon State University
Korea	SNUF	Seoul Univ.		APEX-CE	
	ATLAS	KAERI		OSU-MASLWR	
	VISTA-ITL			NIST	
	FESTA (SMART-ITL)			IST	CAER
				PUMA	Purdue Univ.
				SRI-2	Stanford Research Inst.

+Ref. [59]
**** Available at NEA data bank***

Table A3-2 - ITF simulating PWR, including those considered in NEA/CSNI, 1989, Main characteristics.

Facility	Reference reactor	Scaling approach	Scaling Method	Counter Part Test	ISP-ICSP	Time scale	Volumetric Scale	Height scale	Mass Inventory (kg)	Primary System Volume(m3)	Primary Pressure (MPa)	Core Flow Area (m2)	References
LOFT	W-PWR-4L	TP, RH, FPw, FPr, NC, RLN, S-W	Power-to-Volume (Kv)	YES	05, 09, 11, 13	1	60	0.50		7.63	15.5	0.165	[1]
SEMISCALE	LOFT(MOD 1); W-PWR-4L	TP, FH(Mod 2-3), FPw, FPr, NNC, RLN, S-W	P-to-V (Kv)	YES	02, 04, 08	1	1705	1.00		0.2	15	0.0028	[2]
LOBI	KWU-PWR-4L	TP, FH, FPw, FPr, NNC, RLN, S-W		YES	18	1	712	1.00		0.82	15.5	0.0081	[3]
PKL-III	KWU-PWR-4L	TP, FH, DPw, RPr, NNC, RLN (for PKL I and PKL II), S-W		YES	10 (PKL I)	1	145	1.00		3.3	5	0.042	[4]
LSTF/ (ROSA IV)	W-PWR-4L	TP, FH, DPw, FPr, NNC, RLN, S-W		YES	26	1	48	1.00		7.2	16	0.1134	[5]

CCTF	W-PWR-4L	TP, FH, DPw, RPr, NNC, ELN, S-W				1	21.4	1.00		16	0.6	0.26	[6]
BETHSY	F-PWR-3L	TP, FH, DPw, FPr, NNC, ELN, S-W		YES	27, 38	1	100	1.00		2.9	17.2	0.043	[7]
SPES	W-PWR-3L	TP, FH, FPw, FPr, NNC, ELN, S-W		YES	22	1	427	1.00		0.63	20	0.0096	[8]
OTIS	B&W PWR Raised L - 2×4L	TP, FH, DPw, FPr, NNC, RLN, S-W	P-to-V (Kv) modification of GERDA			1	1686	1.00					[9]
MIST	B&W PWR Lowe. L- 2×4L	TP, FH, DPw, FPr, NNC, ELN, ECLN, S-W	P-to-V (Kv)			1	819	1.00		0.56	15.5	0.0063	[10]
UMCP	B&W PWR Lowe. L - 2×4L	TP, RH Core, FH SG, DPw, RPr, NNC, ELN, ECLN, S-W	See note			1	500	0.33		0.91	2	0.03	[11]
GERDA	B&W PWR Raised L - 2×4L	TP, FH, DPw, FPr, NNC, RLN,	Scaling similar to OTIS			1	1686	1					[12]

		S-W											
SRI-2	B&W PWR - 2×4L	T , RH, RPw, RPr, NNC, ELN, S-W	See note					0.25			0.7		[11]
APEX-CE	CE-PWR - 2L- 1HL 2 CL	TP, RH, RPw, RPr, NNC, ELN, ECLN, S-W	H2TS, Modification of APEX			1	276	0.29		1.12	2.76	0.07	[13]
PWR	**-**	-	-	-	-	**1**	**1**	**1**		**350**	**16**	**4.75**	

SEMISCALE Mod 1 vs. LOFT volume scale is 1/1570.

Cylindrical Core Test Facility (CCTF), Slab Core Test Facility (SCTF) and Upper Plenum Test Facility (UPTF) are part of the 2D/3-D program.

OTIS:A modification of GERDA facility

UMCP: Scaling method adopted is similar to method by Ishii & Kataoka, 1984. Time preserved. Elevations of the various components not preserved.

SRI-2:Scaling method adopted is the method by Ishii & Kataoka, 1984. However, pressure scaling was modified.

Table A3-2A - ITF simulating PWR, including those considered in NEA/CSNI, 1989, Loop and Pump data.

Facility	Number of Loops	Number of Single Loops	Number of Combined intact loops	Single Loop HL ID (mm)	Combined Intact loop HL ID (mm)	Single Loop CL ID (mm)	Combined Intact loop CL ID (mm)	Single Loop Seal Depth (m)	Combined Intact Loop Seal depth (m)	CL N for Loop	Note About Facility Loop	Primary Pump N	Primary Pump Fluid	Specific Speed DN -24260	Primary Pump Coastdown	Note About Primary Pump
LOFT	2	1	3	130	350	130	350	1.27		1		2	1-Phase	Resist.,(64)	I	
SEMISCALE	2	1	3	34	66	34	66	2.7		1		2	2-Phase	16, (22)	I	
LOBI	2	1	3	46	73	46	73	2.1	2.5	1		2	2-Phase	29	P	
PKL-III	4	4	-	128	-	81	-	3.4		1	Preserv. Froude n	4	2-Phase	Speed controlled	C	Note
LSTF/ (ROSA IV)	2	0	2-2	-	207	-	207	-	3.4	1	Preserv. Froude n	2	1-Phase	74.3	C	
CCTF	4	4	-	155	-	155	-	3.4	-	1		0	-	Resistance	-	
BETHSY	3	3	-	118	-	118	-	2.2	-	1	Preserv Froude n	3	2-Phase	28.1	C	
SPES	3	3	-	67	-	67	-	2.9	-	1	Preserv n	3	1-Phase	93.2	P	
OTIS	1	1						9.4		1	Preserv Froude n	0	-	Resistance	-	
MIST	2	2[2×4L]	-	54	-	34	-	9.4	-	2		4	2-Phase	110	P	
UMCP	2	2[2×4L]	-	89	-	76	-	9.4	-	2	Preserv Froude n	0	-	Resistance	-	
GERDA	1									1	Preserv Froude n	0				Note

SRI-2	2	2[2×4 L]	-	52.5	-	40.9	-		-	2		4					centri-fugal pump with vertical axis
APEX-CE	2	2	-	128.2	-	89.9	-		-	2	Preserv. Frouden	4				P	
PWR	**4**	**4**	**-**	**737**	**-**	**737**	**-**	**3.4**	**-**	**1**		**4**	**1-Phase**	**101**		**I**	

(-) refers to combined intact loop geometry

PKL-III: *Pump speed controlled to simulate any pump characteristic*

Gerda: *No pumps are considered in the facility loop, but a multipurpose pump is installed in the CL Bypass*

Primary Pump Coast-down

P: *Programmed*

C: *Controlled*

I: *Inertia*

Table A3-2B - ITF simulating PWR, including those considered in NEA/CSNI, 1989, Core data (see also Karwat, 1985).

Facility	No of (simulated) fuel rods	Diameter (mm)	Pitch (mm)	Length (m)	Maximum Core Power (MW)	Linear Power Rate (average and max. value to be considered)	Heating Method	Power Axial Distribution	Axial Peaking Factor	Note
LOFT	1300	10.7	14.3	1.68	50		**nuclear**			
SEMISCALE	25	10.7	14.3	3.66	2		Indirect		1.58 Mod-1; 1.55 Mod2 to 3	
LOBI	64	10.7	14.3	3.66	5.3		Skin	chopped cosine		
PKL-III	314	10.7	14.3	3.9	2.5		Indirect			
LSTF/(ROSA IV)	1168	9.5	12.6	3.66	10		Indirect	chopped cosine	1.495	
CCTF	2048	10.7	14.3	3.66			Indirect		1.49	
BETHSY	428	9.5	12.6	3.66	3		Indirect			
SPES	97	9.5	12.6	3.66	9		Skin	uniform		
OTIS					0.18					
MIST	45	10.9	14.4	3.66	0.34		Indirect			
UMCP	16	25.4	ca. 80	1.245	0.2		Indirect			
GERDA					0.178					
SRI-2	18	15.9		0.8128	0.088					
APEX-CE	48	25.4		0.9144	0.65			Shaped		
PWR	**51000**	**9.5**	**12.6**	**3.66**	**3800**	**18-45 kW/m**	**nuclear**		**1.495**	**-**

Semiscale Mod 1: Power 1.6 MW (40 rods)
Mod 2 & 3: Power 2.0 MW (25/23 heated rods)

Table A3-2C - ITF simulating PWR, including those considered in NEA/CSNI, 1989, Steam Generator (SG) data.

Facility	Number of SGs	Type (U tube, once-through, horizontal, helical)	U Tubes (Single Loop geometry)	U Tubes (Combined Intact Loop geometry)	Pressure (MPa)	Tube ID / OD (mm)	Pitch(mm)	Average SG U-tube Length (m)	Note
LOFT	1	U tube		1845	6	10.2 / 12.7	19		
SEMISCALE	2	U tube	2	6	6	19.7 / -			
LOBI	2	U tube	8	24	10	19.7/ 22	various		
PKL-III	4	U tube	28		6.0	19.6 / 22	30		
LSTF/ (ROSA IV)	2	U tube	-	141	7.3	19.6 / 25.4	32.5		
CCTF	2	U tube		158	5.2	19.6 / 25.4	32.5		
BETHSY	3	U tube	34	-	8	19.7 / 22	32.5		
SPES	3	U tube	13	-	10	15.4 / 17.5	24.9		
OTIS	1	Once through	19	-	8	14.1 / -			
MIST	2	Once through	19	-	8	14.1 / -			
UMCP	2	Once through	28	-	0.3	30 / 31.7	50.8		
GERDA	1	Once through	19						
SRI-2	2	Once through	48			14.1 / 15.9			
APEX-CE	2	U tube	133	-	2.07	15.42 / 17.45			
PWR	**4**	**U tube**	**3382**	**-**	**6.2**	**19.6 / 22.2**	**32.5**		

Table A3-3 - ITF simulating BWR, including those considered in NEA/CSNI, 1989, Main characteristics.

Facility	Reference Reactor	Scaling Approach	Scaling Method	Counterpart test	ICSP-ISP	Time scale	Volumetric Scale	Height scale	Mass Inventory (kg)	Primary System Volume(m3)	Primary Pressure (MPa)	Core Flow Area (m2)	Jet Pump	REC Loop	References
TLTA	GE-BWR-4 and 6	TP, FH core only, FPw, FPr, NNC, RNJP, ENRL, S-W	P-to-V (Kv)	YES		1	624	1		0.93	7.4	0.0097	2	2	[14]
FIST	GE-BWR/6	TP, FH, FPw, FPr, NNC, RNJP, ENRL, S-W		YES		1	624	1		0.67	7.4	0.0116	2	2	[15]
ROSA-III	GE-BWR/6	TP, RH, RPw, FPr, NNC, RNJP, ENRL, S-W		YES	ISP 12	1	424	0.5		1.42	7.2	0.0392	4	2	[16]
TBL	GE-BWR/5	TP, FH, FPw, FPr, NNC, RNJP, ENRL, S-W					350	1		1.6	7.2	0.0232	2	2	[17]

FIX-II	AA-BWR	TP, FH, FPw, FPr, NNC, S-W			ISP 15	1	777	1		0.46	7.4	0.006	Ext. Pumps	2	[18]
Piper-1	GE-BWR-4 and 6	TP, FH, DPw, FPr, NNC, RNJP, NORL, S-W		YES	ISP 21	1	2200	1		0.19	7.4	0.0028	1	None	[19]
DESIRE	Dodewaard NC BWR	RH, NNC, Freon-12	*					0.5			1.3				[20]
CIRCUS	Dodewaard NC BWR	FH, RPr, NNC, S-W						1			0.5				[20]
BWR	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**620**	**7.8**	**8.6**	**24**	**2**	

TLTA: *Predecessor of FIST. The jet pumps are linearly scaled to the height and diameter. In FIST these are modified to full height.*

DESIRE: *pressure range: 8 -13 bar*

CIRCUS: *pressure range: 1-5 bar*

CIRCUS: *riser diameter is 47mm*

* *see also Van De Graaf et al., 1994a.*

Table A3-3A - ITF simulating BWR, including those considered in NEA/CSNI, 1989, Core data.

Facility	No of (simulated) fuel rods	Diameter (mm)	Pitch (mm)	Length (m)	Maximum Power (MW)	Linear Power (average and max. values to be considered)	Heating Method	Axial Power Distribution	Axial Peaking Factor	Rod Surface Temperature	Note
TLTA	64	12.3	16.2	4.14	7		Skin				
FIST	64	12.3	16.2	4.14	7		Skin	chopped cosine	1.4		
ROSA-III	284	12.3	16.2	1.88	4.4		Indirect	chopped cosine	1.4		
TBL	128	12.5	16.2	3.7	10		Indirect				
FIX-II	36	12.3	16.3	3.7	3.5		Skin		1.21		
Piper-1	16	12.3	16.2	4.3	0.25		Indirect	cosimusoidal	1.26		
DESIRE	35	6.35		0.88	0.05			chopped cosine, flat uniform			
CIRCUS	4	12.5			0.012						glass channel
BWR	**52576**	**12.3**	**16.2**	**3.705**	**3800**	**18-45 kW/m**	**nuclear**		**1.4**		

CIRCUS: maximum power per rod is 3 kW

Table A3-4 - ITF and SETF simulating VVER, considered in NEA/CSNI, 2001, Main characteristics.

Facility	Reference Reactor	Scaling Approach	Scaling Method	ISCP-ISP	Time scale	Volumetric Scale	Height scale	Mass Inventory (kg)	Primary System Volume(m3)	Primary Pressure (MPa)	Core Flow Area (m2)	Refernces
PACTEL	VVER 440-6L	TP, FH, RPw, RPr, NNC, RLN, S-W	P-to-V (Kv)	ISP 33	1	305	1			8		[21]
PMK-2	VVER 440-6L	TP, FH, FPw, FPr, NNC, RLN, S-W		Test IAEA-SPE4	1	2070	1			12.35		[22]
REWET-III	VVER 440-6L	FH, RPw, RPr, NNC, RLN, S-W	Note			2333	1			0.35		[23]
KMS		FH reactor scale, R scale cont., FPr, S-W				27	1			18		[23]
PSB	VVER 1000-4L	TP, FH, FPw, FPr, NNC, ELN, S-W	P-to-V (Kv)		1	300	1			20		[24]
PM-5	VVER 1000-4L	RA, RH, RPr, RLN, S-W					0.2			0.3		[23]
SB	VVER 440 -6L	FH, RPw, FPr, RLN, S-W				300	1			16		[23]
ISB	VVER 1000-4L	TP, FH, FPw, FPr, NNC, RLN, S-W	P-to-V (Kv)	Test UPB-2.4 is first Russian Standard Problem	1	3000	1			25		[23]
BD	VVER 1000-4L	RPr, RLN, S-W				5				1		[23]
VVER 1000					**1**	**1**	**1**			**15.7**		
VVER 440					**1**	**1**	**1**			**12.26**		

REWET: for natural circulation (NC) test:pressure is 1 bar;
For compensated leak test (single-phase NC)pressure is 0.1-0.35 MPa

REWET-III: Modified from REWET II for two-phase natural circulation studies

KMS:	*Containment linear scale is 1:3 (D: 6.0 m; H: 20 m; V: ~2000m3; P up to 0.6 MPa)*
PM-5:	*Pressure: 1-3 bar (1-1.5 bar for 'transparent' model)*
BD:	*One circulation loop with scale 1:5 loop seal and circulator & three working loops without circulator (see also Table A3-4A)*
ISB:	*Test UPB-2.4 provided data for the first Russian Standard Problem*
SB:	*It models also a VVER 1000-4Loops with a volumetric scale of 1/3000 and full height*

Table A3-4A - ITF and SETF simulating VVER, considered in NEA/CSNI, 2001, Loop and Pump data.

Facility	Number of Loops	Number of Single Loops	Number of Combined Intact Loops	Single Loop HL ID (mm)	Combined Intact Loop HL ID (mm)	Single Loop CL ID (mm)	Combined Intact Loop CL ID (mm)	Single Loop Seal Depth (m)	Combined intact loop Seal depth (m)	CL N for Loop	Loop Data Note	Primary Pump N	Primary Pump Fluid	Spec. Speed DN -24260	Primary Pump Coastdown	Primary Pump Note
PACTEL	3	-	2	-	52.5	-	52.5	-		1	prese rv Froud e n	3				Primary pumps simulated by flow resistances (pumps will be added later)
PMK-2	1	-	6	-	46	-	46	-		1	prese rv Froud e n	1				Pump is accommodated in by-pass line: flow rate 0 to nominal value, NPP coast down simulation
REWET-III	1	1	-	22	-	22	-			1						In REWET II test facility, SG and primary circulating pumps were simulated by using flow resistance.
KMS	4															
PSB	4	4	-	76	-	76	-			1		4				Primary pumps installed in each loop
PM-5	1			60		60										Pump: 5000 kg/h
SB	2															Intact loop contains main circulation pump, heater and cooler.
ISB	2	1	3	25	56	25	41			1		2				Primary pumps installed in bypasses of each loop

BD	1															One circulating loop with loops seal and circulator. Three working loops without circulator
VVER 1000	**4**	**4**	**-**	**850**	**-**	**850**	**-**			**1**		**4**				
VVER 440	**6**	**6**	**-**	**496**	**-**	**496**	**-**			**1**		**6**				

ISB: Intact loop includes three separate SGs

BD: Three loops are not entirely modeled

Table A3-4B - ITF and SETF simulating VVER, considered in NEA/CSNI, 2001, Core Data.

Facility	No	Dia (mm)	Pitch (mm)	Length (m)	Maximum Power (MW)	Linear Power (average and max. values to be considered)	Heating Method	Axial Power Distribution	Axial Peaking Factor	Rod Surface Temperature	Note
PACTEL	144	9.1	12.2	2.42	1		Indirect	chopped cosine	1.4		
PMK-2	19	9.1	12.2	2.5	0.664			Uniform axial			
REWET-III	19	9.1	12.2	2.4	0.03			chopped cosine			
KMS		9.15		3.53	30		Direct and indirect				
PSB	168	9.1	12.75	3.53	15		Indirect				
PM-5	70				0.35						
SB		9.1	12.75	3.5	1		Indirect	Uniform & Stepwise			
ISB	19	9.1	12.75	3.53	1.8		Direct and indirect				
BD											
VVER 1000	**50856**	**9.1**	**12.75**	**3.53**	**3000**		**Nuclear**				
VVER 440	**39312**	**9.1**	**12.2**	**2.42**	**1375**	**12.5 -32.5 kW/m**	**Nuclear**				

REWET: 19 rod bundle;

KMS: 2184 rods (all bundles);

SB: 7 or 19 rod bundles.

Table A3-4C - ITF and SETF simulating VVER, considered in NEA/CSNI, 2001, Steam Generator (SG) data.

Facility	Number of SGs	Type (vertical, Horizontal..)	U Tubes (Single Loop geometry)	U Tubes (Combined Intact Loop geometry)	Pressure (MPa)	ID (mm)	OD (mm)	Pitch(mm)	Average SG U-tube Length (m)	Note
PACTEL	3	Horizontal SG	-	38	4.65	13			8.8	
PMK-2	1	Horizontal SG	-		4.6					
REWET-III	1	Horizontal SG	12	-		13				
KMS					up to 12 Mpa					
PSB	4	SGs of special design eight slightly inclined full length tubes	34	-	13	16				
PM-5		SG - is not modelled (for full height model SG incorporates 20 full length tubes								
SB	-	SG are modeled by coolers in each loop								
ISB	4	Vertical U tubes	11	3*11	13	11	16		2.71	
BD										
VVER 1000	**4**	**Horizontal**	**11000**	**-**	**7.9**	**13**	**16**		**11.1**	
VVER 440	**6**	**Horizontal**	**5536**	**-**	**4.6**	**13,2**	**16**		**9.02**	

ISB: the intact loop includes 3separate SGs.

Table A3-5 - ITF simulating advanced PWR,

Main characteristics.

Facility	Reference Reactor	Scaling Approach	Scaling Method	Counterpart /similar Test	ICSP-ISP	Time scale	Volumetric Scale	Height Scale	Mass Inventory (kg)	Primary System Volume(m3)	Primary Pressure (MPa)	Core Flow Area (m2)	References
PWR PACTEL	EPR Like-4L	RPw, RPr, NNC, RNL, S-W					RPV: 1:405 SG: 1:400 PRZ: 1:565	RPV(core): 1:1, SG: 1:4, PRZ: 1:1.6			8		[25]
ATLAS	APR1400 - 2L (1HL-2CL)	TNP, RH, RPw, FPr, NNC, ELN, ECLN, S-W	Three-level scaling	YES		0.71	288	0.5		0.55	20		[26, 27]
SNUF	APR1400 - 2L (1HL-2CL)	RH, RPw, RPr, NNC, ELN, ECLN, S-W	Three-level scaling				1139.2	0.16		0.23	0.8		[28, 29]
APEX	AP600-2L (1HL-2CL)	TNP, RH, RPw, RPr, NNC, ELN, ECLN, S-W	H2TS	YES		0.5	192	0.25			2.76		[30,31]
SPES-2	AP600-2L (1HL-2CL)	TP, FH, FPw, FPr, NNC, ELN, ECLN, S-W	P-to-V (Kv) Modification of SPES	CT, ST		1	395	1			20	0.0096	[32]
ROSA-AP600	AP600-2L (1HL-2CL)	TP, FH, RPw, FPr, NNC, ELN, NCLN, S-W	P-to-V (Kv) Mmodification of LSTF	YES		1	30.5	1			16	0.1134	[33]
OSU-MASLWR	MASLWR - IWCR	TP, RH, FPw, FPr, NNC, I, S-W-A	H2TS		IAEA ICSP	1	254.7	0.33			11.4	0.0084	[34]

VISTA-ITL	SMART-IWCR	TNP, RH, FPw, FPr, NNC, S-W	Three-level scaling			0.6	1310	0.36		0.2233	17.2	0.00165	[35]
FESTA	SMART-IWCR	TP, RPw, FPr, NNC, S-W	Three level scaling			1	49	1		5.0616	18	0.03	[36]
IST	mPOWER IWCR	TP, FH, FPr, NNC, S-W				1		1					[37]
EPR	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**455**	**15.5**		
APR1400	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**454.7**	**15.5**		
AP600	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**239**	**15.5**		
MASLWR	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**			**8.6**		
SMART	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**56.27**	**15**		
PWR	**-**	**-**	**-**	**-**	**-**	**1**	**1**	**1**		**350**	**16**	**4.75**	

OSU-MASLWR: Upper region of the hot leg riser has an OD equal to 114.3 mm;

NIST: A modification of the OSU-MASLWR facility and used for the simulation of NUSCALE-IWCR and TP, RH, FPw, FPr, NNC, I, S-W-A are the main characteristics

CT: Counterpart Test

ST: Similar Test

Table A3-5A - ITF simulating advanced PWR,

Loop and Pump data.

Facility	Number of Loops	Number of Single Loops	Number of Combined Intact loops	Single Loop HL ID (mm)	Combined Intact Loop HL ID (mm)	Single Loop CL ID (mm)	Combined Intact Loop CL ID (mm)	Single Loop Seal depth	Combined Intact Loop Seal Depth (m)	CL N for Loop	Loop Note	Primary Pump N	Primary Fluid	Spec. SpeedDN -24260	Primary Pump Coast down	Note
PWR PACTEL	2	2	-	52.5	-		-		-	1		0				
ATLAS	2	2	-	131.8	-	87.3	-		-	2	preserv Frouden	4			P	
SNUF	2	2	-	51	-	64	-		-	2		4				
APEX	2	2	-		-		-		-	2	preserv Frouden	4			P	Simulation of the canned motor pump of AP600. Attached at the lower channel head of the SG
SPES-2	2	2	-		-		-		-	2	preserv Frouden	2	1-phase		P	centrifugal-single stage horizontal shaft type
ROSA-AP600	2	2	-		-		-		-	1	preserv Frouden	2	1-phase		C	
OSU-MASLWR	1	1	-		-		-	-	-	-		-	-		-	Integral design natural circulation facility
VISTA-ITL	1	1	-	42.9	-	53.97	-	-	-	1	preserv Frouden	1	1-phase		C	simulation of the canned motor pump of SMART
FESTA	4	4	-	110	-	110	-	-	-	1	preserv Frouden	4	1-phase		C	simulation of the canned motor pump of SMART
IST																

EPR	4	4	-		-		-		-	1		**4**				**Single-stage, centrifugal pump**
APR 1400	**2**	**2**	**-**	**1068**	**-**	**762**	**-**		**-**	**2**	**-**	**4**				**Vertical, single-stage, centrifugal pump**
AP600	**2**	**2**	**-**					**-**	**-**	**2**		**4**				**Canned motor**
MASLWR												**-**				Integral design natural circulation reactor
SMART												**4**				**Canned motor, Axial**
PWR	**4**	**4**	**-**	**737**	**-**	**737**	**-**	**3**	**-**	**1**		**4**	**1-Phase**	**101**	**I**	

P: Programmed, C: Controlled, I: Inertia

Table A3-5B - ITF simulating advanced PWR,

Core data.

Facility	No of (simulated) Fuel Rods	Rod Diameter (mm)	Pitch (mm)	Length (m)	Maximum Power (MW)	Linear Power (average and max. values to be considered)	Heating Method	Axial Power Distribution	Axial Peaking Factor	Note
PWR PACTEL	144			2.42	1			Chopped cosine		
ATLAS	396	9.5	12.85	1.905	2		Indirect	Chopped cosine	1.466	
SNUF	260			0.6	0.27					
APEX	48	25.4		0.914	0.6			Shaped		
SPES-2	97	9.5	12.6	3.66	9		Skin heated	Uniform		
ROSA-AP600	1008	9.5	12.6	3.66	10		Indirect	chopped cosine	1.495	
OSU-MASLWR	57	15	18.6	0.686	0.6					
VISTA-ITL	36	9.5	11.3	1.2	0.8188		Indirect	-	-	
FESTA	304	9.5	12.6	2	3		Indirect	chopped cosine	1.509	
IST										
EPR	**63865**	**9.5**		**4.2**	**4250**	**14.95-**	**nuclear**			
APR 1400	**56876**	**9.5**		**3.75**	**4000**		**nuclear**			
AP600	**38280**	**9.5**		**3.658**	**1940**	**13.5 -**	**nuclear**			
MASLWR	**6336**	**9.5**	**12.6**	**1.35**	**150**		**nuclear**			
SMART	**15048**	**9.5**		**2**	**330**	**11.9-**	**nuclear**			
PWR	**51000**	**9.5**	**12.6**	**3.66**	**3800**	**18-45 kW/m**	**nuclear**		**1.495**	**-**

Table A3-5C - ITF simulating advanced PWR,

Steam Generator (SG) data.

Facility	Number of SGs	Type (U tube, once-through, horizontal, helical)	U Tubes (Single Loop geometry)	U Tubes (Combined Intact Loop geometry)	Pressure (MPa)	ID (mm)	OD (mm)	Pitch(mm)	Average SG U-tube Length (m)	Note
PWR PACTEL	2	U Tube	51	-	5	16.6	19.05	27.4	6.5	
ATLAS	2	U tube	176	-	10	12	14.2	20	9.16	
SNUF	2	U Tube	16	-						
APEX	2	U tube		-						
SPES-2	2	U tube	13	-	10	15.4	17.5	24.9		
ROSA-AP600	2	U Tube	141	-	7.3	19.6	25.4	32.5		
OSU-MASLWR	1	Helical	14	-	1.5	12.6	15.9			
VISTA-ITL	1	Helical	12	-	17.2	7	10			
FESTA	4	Helical	15	-	18	12	17			
IST										
EPR	**4**	**U tube**	**5980**		**10**	**16.87**	**19.05**			
APR 1400	**2**	**U tube**	**12596**	**-**	**6.9**	**16.9**	**19.05**		**19.96**	
AP600	**2**	**U tube**	**6307**	**-**	**5.74**	**15.5**	**17.5**			
MASLWR	**1**	**Helical**	**1012**	**-**	**2.1**	**14.2**	**16**			
SMART	**12**	**Helical**	**324**	**-**	**3.0**					
PWR	**4**	**U tube**	**3382**	**-**	**6.2**	**19.6**	**22.2**	**32.5**		

Table A3-6 - Facilities constructed and operated for the simulation of containment including advanced designs.

Country	ITFs	Organization
Australia	Lucas Heights blowdown/ containment test rig	Australian Atomic Energy Commission
Canada	SSBT	AECL
Czech Republic	SVUSS	SVUSS & GRS
France	MISTRA	CEA
Germany	HDR	KfK/BMFT
	BFC (or BFMC or BMC)	Battelle lngenieurtechnik GmbH or Battelle-Institut e.V
	THAI	Becker Technologies GmbH
	PSS	GKSS
	GKM I & II	KWU
	INKA	*AREVA NP GmbH*
Japan	-	JAERI
	JAERI Full Scale Mark II	
Russia	BC-V-213	EREC
Sweden	Marviken*	Studsvik
USA	CVTR	Carolinas Virginia Nuclear Power Associates, Incorporated
	CSTF	HEDL
	PSTF	GE
	4T	
	FSTF	
	PCCS Large scale test facility	Westinghouse Science and Technology Center

Available at NEA Data Bank

Table A3-7 (1/2) - CONTAINMENT facilities including those considered in NEA/CSNI, 1989a, and NEA/CSNI, 1999 – Main characteristics (1/2)

Facility	Ref Reactor	Scaling	ISP	Shape	Pressure Bondary Type	Pressure (kPa(a))	Volume (m3)	Height (m)	Diameter (m)
HDR	Existing Cont.	No specific scaling consideration	ISP16, 23, 29	C	ST	600	11300	60	20.7
BFC		E. Input / Cont V. = cost	CASP 1 CASP 2	C	CON	~500	640	9	12
CVTR	Existing Cont.	Not known		C	CON	150	6430	34.7	17.373 6
Australian experiment			CASP 3	C	ST		1.81	3.048	0.914
CSTF	Ice Condenser Containme nt	Linear scaling: 0.3		C	ST	~500	850	20.3	7.6
AP600 PCCS	AP600	Note		C	ST	689	88	6.1	4.6
MISTRA	French PWR Cont	Linear Length scale:0.1	ISP 47	C	ST	~600	99.5	7.3	4.25
THAI	-	-	ISP 47	C	ST	1400	60	9.2	3.2
German PWR Containme nt	**-**	**-**	**-**	**S**	**ST**	**530**	**70000**	**56**	**56**
US PWR Containme nt	**-**	**-**	**-**	**C**	**STL C**	**420-520**	**77000**	**64**	**43**
Japanese PWR Containme nt	**-**	**-**	**-**	**C**	**STL C**	**~500**	**72900**	**65**	**43**
French PWR Containme nt	**-**	**-**	**-**	**C**	**CON**	**530**	**73050**	**63**	**44**
CANDU 6 Containme nt	**-**	**-**	**-**	**C**	**CON**	**225**	**48000**	**46**	**41.7**

MISTRA: Characterized by Stainless Steel material

THAI: *Characterized by Stainless Steel material; the facility could be divided into 5 compartments; the medium that could be used are the steam, water, air, hydrogen and helium; it is characterized by heat losses of 8 kW at 100°C*

Another vessel has been added and the extended THAI test facility could be sub-divided into more than five compartments on demand

AP600 PCCS: *One-eighth scale model of AP600 containment with a prototypic height-to-diameter ratio.*

Shape: C: Cylindrical, S: Spherical,

Pressure Boundary Type: CON: concrete, ST: Steel, STLC: Steel lined-concrete

Table A3-7 (2/2) - CONTAINMENT facilities including those considered in NEA/CSNI, 1989a, and NEA/CSNI, 1999 – Main characteristics (2/2)

Facility	Dome shape	Dome Vol(m3)	Compartment	LT heat sink S/V ratio	Steel (including cont. shell) (m2)	Concrete (m2)	ToLal Surfaces (m2)	Surface/Volume (m-1)	References
HDR	H	4800	~70	0.74	21000	8400	29400	2.6	[43]
BFC	F	~260	9	1.6	125	1020	1135	1.77	[43]
CVTR	H		3	0.20	2400	1300	3700	0.58	[44]
Australian experiment			2		13.89		13.89	7.67	[45]
CSTF	E	~650	2	0					[46]
AP600 PCCS	E	~70	1-4	0					[47]
MISTRA	F			0					[48]
THAI	H	17.7	M	0	163	0	163	2.72	[49]
German PWR Containment	**H**	**41000**	**120**	**0.44**	**14500**	**30700**	**45200**	**0.65**	
US PWR Containment	**E**	**~60000**	**20**	**0.25**					
Japanese PWR Containment	**H**	**52000**	**~25**	**0.25**					
French PWR Containment	**E**	**~47000**	**~50**	**?**					
CANDU 6 Containment	**E**	**~27000**	**~20**	**0.4**					

Long-term heat sink's surface-to-volume ratio (concrete and steel-clad concrete walls only)

Long term heat sink is due to concrete heat structures; Short term heat sinks is due to steel internal structures

Dome Shape: H: Hemispherical, F: Flat, E: Elliptical,

Compartment: M: Multi-compartment,

Table A3-8 - Pressure suppression containment systems, NEA/CSNI, 1986, comparison of Selected data

	Parameter	GE-MARKI USA	GE-MARKII USA	GE-MARK III USA	MARK I Japanese Modification	MARK II Japanese Modification	KWU-69 FRG	KWU 72 FRG	External Pumps Sweden	Internal Pumps Sweden
Drywell	Power level (MWt)	3300	3300	3800	3300	3300	3800	3800	1800	3000
	Volume (m3)	4500	5700	7900	8800	8700	5000	8500	5000	5800
	Design pressure (kPa)	530	410	270	410	410	540	430	500	600
	Working diamter (m)	20	26	22	24	29	30	29	22	25
Wetwell	Water volume (m3)	3300	3100	4100	5300	5700	3700	3100	1900	3200
	Air Volume (m3)	4500	4100	33000	3800	4000	2700	6000	3000	2900
	Design pressure (kPa)	530	410	200	410	410	540	430	500	200
	Design temperature(°C)	135	85	85	n.a	n.a	n.a	n.a	95	95
Vents	Orientation	vertical	vertical	horizontal	vertical	vertical	vertical	vertical	vertical	vertical
	Dia (m)	0.6	0.6	0.7	0.6	0.6	0.6	0.6	0.6	0.6
	Area (m2)	21	30	43	23	30	20	18	27	7

Table A3-9 - Pressure suppression containment facilities, NEA/CSNI, 1986, comparison of selected geometrical data.

			Pipes				Volume (m^3)			
Facility	**Reference Containment Type**	**Vent Type**	**Number of Vent Pipes**	**Diameter (m)**	**Vent Area (m^2)**	**Top Vent Submergence (m)**	**Dry Well**	**Air Wet Well**	**Pool**	**Pool Surface Area (m^2)**
PSTF	BWR6 MARK III	Straight horizontal	3	0.7	1.15	0.6-4.7	67 (124)	300	70	12.4
			3	0.4	0.38	1.6-3.1	67	300	24	4.30
			9	0.2	0.37	1.6 - 3.1	67	open	24	4.27
4T	BWR 4 MARK II	Straight vertical	1	0.5 to 0.6	0.19 to 0.27	2.7-4.1	53	26 to 31	20 to 25	3.20
FSTF	BWR 2 -3 Mark I	?	8	0.6	2.3	0.5-1.4	237	260 to 300	180 to 220	47.4
MARKIVEN		Straight vertical	58*	0.3	4.03*	2.8	1980	560	1580	110
CRT	MARK II	Straight vertical	4-7	0.6	1.91	3.3-3.9	329*	255	187	24.8
PSS	SWR 69	Straight vertical	3	0.6	0.88	2.8*	59.8	47.5 (72.5)	64.1	16.2
GKM	SWR 69	Straight vertical	1	0.6	0.29	n a	n a	n a	n a	n a

** can be changed as a parameter*

Table A3-10 - Pressure suppression containment facilities, NEA/CSNI, 1986, key scaling factors.

Facility	Drywell Volume	Air Wetwell volume	Pool Volume	Height	Blow Down Flow Area /Vent Pipe Flow Area	WW Pool Surface Area / Vent Pipe Flow Area	DW Volume /BD Flow Area (m)	DW Volume /WW Air Volume	Reference
PSTF	1/128 (1/69)	1/112	1/46	1/1	0.0027	10.7	2.1 x10^4	0.21-0.36	[50]
	1/128	1/112	1/134	1/1	0.0027	11.30	2.1 x10^4	0.21-0.36	
	1/128	1/112	1/135	1/1	0.0027	11.5	2.1 x10^4	0.21-0.36	
4T	1/110	1/110	1/110	1/1	0.012 to 0.024	11.6 to 17.4	1.17 x10^4 to 1.17 x10^4	1.17-2.0	[51]
FSTF	1/16	1/16	1/16	1/1	0.0036 to 0.018	20.3	0.6 x10^4 to 2.8 x10^4	0.8-0.9	[52]
MARKIVEN	1/1	1/1	1/1	1/1	0.069	27.3	3.2 x10^4	3.5	[53]
CRT	1/18	1/16	1/18	1/1	0.024	13.0*	> 7.27 x10^3 *	1.29*	[54]
PSS	1/95	1/86 (1/57)	1/48	1/1	0.009	18.4	7.6 x10^3	1.26	[55]
GKM	1/110	na	na	1/1	na	na	na	Na	[56]

** Can be changed as a parameter*

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ACRONYMS AND ABBREVIATIONS for Appendix A-3

(In addition to the Acronyms and Abbreviations used for the Main Text and for A-1, A-2 and A-2)

AECL	Atomic Energy of Canada Limited
APEX	Advanced Plant Experiment (AP600 test facility)
APEX-CE	Advanced Plant Experiment – Combustion Engineering
ATLAS	Advanced Thermal-hydraulic Test Loop for Accident Simulation
BC	Bubble Condenser
BD	Russian test facility
BETSHY	Boucle d'Etudes Thermohydrauliques Système
BFC	Battelle-Frankfurt Containment experiment
BMWi	Bundesministerium für Wirtschaft und Technologie
C	Controlled (Primary Pump Coast down)
C	Cylindrical (Containment Shape)
CCTF	Cylindrical Core Test Facility (2D/3-D)
CEA	Commissariat à l'Energie Atomique et aux Energies Alternatives
CON	Concrete (Containment Pressure boundary type)
CSTF	Containment Systems Test Facility
CT	Counterpart test
CVTR	Carolinas Virginia Tube Reactor Containment
DPw	Decay Power
E	Elliptical (Containment Dome Shape)
ECLN	Equal CL number
ELN	Equal Loop Number
ENRL	Equal Number Recirculation Loop
ESBWR	Economic Simplified Boiling Water Reactor
F	Flat (Containment Dome Shape)
FESTA	Facility for Experimental Simulation of Transients and Accidents (namely SMART-Integral Test Loop)
FH	Full Height
FIST	Full Integral Simulation Test
FIX-II	Swedish BWR - Related Test Facility
FPr	Full Pressure
FPw	Full Power
FSTF	Full-Scale test facility

GE	General Electric Company
GERDA	Geradrohr Dampferzeuger Anlage
GIRAFFE	Gravity-Driven Integral Full Height Test for Passive Heat Removal
GIST	GDCS (Gravity Driven Cooling System) Integral System Test
GKM	Grosskraftwerk Mannhein AG
GKSS	Gesellschaft fur Kernenergie in Schiffsbau u. Schiffahrt
H	Hemispherical (Containment Dome Shape)
HDR	HeiBdampfreaktor
HEDL	Hanford Engineering Development Laboratory
I	Inertia (Primary Pump Coast down)
I	Integral design
INKA	Integral Test Facility Karlstein
IRSN	Institut de Radioprotection et de Sûreté
ISB-VVER	Russian Test Facility
ISP	International Standard Problem
IST	Integrated Systems Test (IST) Facility
IVO	Imatran Voima Oy
JAEA	Japan Atomic Energy Agency
JAERI	Japan Atomic Energy Research Institute
JRC	Ispra Joint Research Centre of the European Commission, Ispra, Italy
KMS	Russian Test Facility
KWU	Kraftwerk Union AG
KAERI	Korea Atomic Energy Research Institute
LOBI	Loop for Blowdown Investigation or Loop for Off-normal Behavior Investigations
LOFT	Loss-of-Fluid-Test
MASLWR	Multi-Application Small Light Water Reactor
MIST	Multi-loop Integral System Test Facility
MISTRA	Mitigation and STRAtification facility
M	Multi-compartment (Containment)
NC	Nuclear Core
NNC	Non-Nuclear Core
NIST	NuScale Integral System Test
NUPEC	Nuclear Power Engineering Corporation
OSU	Oregon State University
OTIS	Once-Through Integral System
P	Programmed (Primary Pump Coast down)
PANDA	Passive Nachwärmeabfuhr- und Druckabbau-Testanlage
PACTEL	Parallel Channel Test Loop
PCCS	Passive Containment Cooling System
PIPER-ONE	Italian BWR-related Test Facility
PKL	Primärkreislauf
PM-5	Russian Test Facility
PMK-2	Hungarian Test Facility
PSB-VVER	Russian Test Facility
PSTF	Pressure Suppression Test Facility
PUMA	Purdue University Multi-dimensional Integral Test Assembly
ROSA	Rig of Safety Assessment
ROSA-LSTF	ROSA- Large Scale Test Facility
REWET	Finnish Test Facility
RH	Reduced Height
RL	Recirculation Loop

RLN	Reduced Loop Number
RNJP	Reduced Number of Jet Pump
RPr	Reduced Pressure
RPw	Reduced Power
S	Spherical (Containment Shape)
SB	Small Break; Russian Test Facility
SCOP	SMART Core flow distribution and Pressure drop test facility
SCTF	Slab Core Test Facility (2D/3-D)
SMART	System-Integrated Modular Advanced ReacTor
SNUF	Seoul National University Facility
SPES	Simulatore PWR per Esperienze di Sicurezza
SRI	Stanford Research Institute
SSBT	Small Scale Tube Burst Facility
ST	Similar Test
ST	Steel (Containment Pressure Boundary Type)
STLC	Steel lined-concrete (Containment Pressure Boundary Type)
SVUSS	Statni Vyzkumny Ustav pro Stavbu Stroju
S-W	Steam – Liquid Water
4T	Temporary Tall-Tank Test
TBL	Two Bundle Loop
ThAI	Thermal-hydraulics, Hydrogen, Aerosols, Iodine
TLTA	Two-loop Test Apparatus
TNP	Time Not Preserved
TP	Time Preserved
UMCP	University of Maryland College Park (UMCP) 2×4 loop
VISTA-ITL	Experimental Verification by Integral Simulation of Transients and Accidents – Integral Test Loop
VTT	Technical Research Centre of Finland

A-4 AN OUTLINE OF CODE DEVELOPMENT AND V & V

Appendix 4 consists of three parts (A-4.1, A-4.2 and A-4.3) dealing with code development, verification and validation, respectively.

A-4.1: The process of code model development

SYS TH codes model the thermal-hydraulic physical system and related coupled other systems. The thermal-hydraulic system can be either the cooling circuits of a nuclear reactor, or the circuits of a test facility, to be simulated by solving systems of equations. The thermal-hydraulics of the cooling circuits is generally treated by a generic method used for all components. However some specific components having a particular geometry require some specific thermal-hydraulic models. Thermal-hydraulics is also coupled to non- thermal-hydraulic systems which are also modeled in SYS TH codes. The following subsections give an overview of the generic thermal-hydraulic model, some specific models, and the non-thermal-hydraulic systems.

A-4.1.1 Fundamental models for thermal-hydraulics

The thermal-hydraulic equations express the basic principles of the Physics:

- Mass conservation and energy conservation
- Newton law,
- Second principle of thermodynamics

These are physical laws which are universally valid.

In Newtonian fluids, these physical laws are expressed through the continuity equation, the Navier Stokes equation and the energy equation which can be a transport equation for the internal energy u, or the total energy u+v2/2, or the enthalpy h, or the total enthalpy h+v2/2, or even the entropy.

$$\frac{\partial \rho}{\partial t} + \frac{\partial \rho v_j}{\partial x_j} = 0$$

$$\frac{\partial \rho v_i}{\partial t} + \frac{\partial \rho v_i v_j}{\partial x_j} + \frac{\partial p}{\partial x_i} = \rho g_i - \frac{\partial \sigma_{ij}}{\partial x_j}$$

$$\frac{\partial \rho u}{\partial t} + \frac{\partial \rho v_j h}{\partial x_j} - v_j \frac{\partial p}{\partial x_j} = \rho g_i v_i - \frac{\partial \sigma_{ij} v_j}{\partial x_j} + \frac{\partial q_j}{\partial x_j}$$

However in fluids these physical laws are applied with some approximations due to space and time averaging. This averaging allows to focus on some macroscopic phenomena of interest and to avoid spending an unaffordable CPU time in solving all the microscopic details of the flows. However this averaging also induces some approximations to exact equations and the solution of these averaged equations is also an approximate solution of the initial problem.

Since these averaged equations simplify the physical reality, they are no more exact equations, they are called models. After averaging of local instantaneous basic equations, many additional terms appear in the balance equations which require closure relations or constitutive relations.

Cooling fluid in Light Water Reactors is either liquid water, steam, or a two-phase steam-water mixture. Since two-phase flow may exist in all components of both Pressurized water Reactors and Boiling Water Reactors either in normal or accidental situations, thermal-hydraulic equations are written for a two-phase mixture including possible non condensable gases. Historically the description of two-phase flow was made with an increasing level of detail. The first level was the Homogeneous Equilibrium Model (HEM), followed by drift flux model, and finally the two-fluid model that is used in most current system codes, and some multi-field models were developed either for component codes (codes which simulate only one component of the reactor such as a Core, or a Steam Generator, or a Heat Exchanger) or for system codes. All these models use a double space and time averaging of basic equations.

Time averaging allows filtering all fluctuations due to turbulence and to the two-phase intermittency (succession of liquid and gas phases at a given point). The time scale of the averaging is supposed to be larger than the largest scale due to turbulence and to the two-phase intermittency and smaller than the time scale of variation of the averaged flow parameters during transients.

$$\bar{F}(x,t) \triangleq \frac{1}{\Delta t}\int_{t-\Delta t/2}^{t+\Delta t/2} F(x,t)dt$$

$$F = \bar{F} + F'$$

$$\alpha_k \triangleq \overline{\chi_k}$$

In fact the following time averaging of flow parameter F is used for each phase k:

$$\bar{F}^k(x,t) \triangleq \frac{1}{\alpha_K \Delta t}\int_{t-\Delta t/2}^{t+\Delta t/2} \chi_k\, F(x,t)dt$$

The space averaging may be:

– a volume averaging: used in the control Volume context or in 3-D Pressure Vessel Modules in the frame of a porous body approach
– a cross section averaging: used in 1-D modules for pipes, ducts, channels
– an averaging over one space dimension: e.g. for a 2-D modelling of an annular downcomer the equations are averaged over the radial dimension from Pressure Vessel wall to the core barrel

The space averaging allows a macroscopic description with a relatively coarse space resolution and requires relatively short computer time to simulate complex transients.

Before any averaging, local instantaneous equations may be multiplied by fluid-solid characteristic functions (in the frame of a porous 3-D modelling), by phase characteristic functions (in the frame of the two-fluid model), or field-characteristic functions (in the frame of the multi-field model).

Let $\chi_f(x,t)$ be the fluid/solid characteristic function:

$\chi_f(x,t)$= 1 when point x is in the fluid at time t

$\chi_f(x,t)$= 0 when point x is in the solid at time t

In case of a flow bounded by non-deformable solid structures, $\chi_k(x)$is not function of time.

A Volume average of A is defined as:

$$\langle F(x,t)\rangle \triangleq \frac{1}{V(x)}\int_{V(x)} F(y,t)dV(y)$$

A Volume average of χ_f is the so-called porosity factor:

$$\phi \triangleq \langle \chi_f \rangle = \frac{V_f}{V}$$

In the classical porous body approach, after multiplication by χ_f , equations are averaged over a fluid volume as follows:

$$\langle F(x,t)\rangle_f \triangleq \frac{\langle \chi_f F\rangle}{\langle \chi_f \rangle} \triangleq \frac{1}{V_f(x)}\int_{V(x)} \chi_f \, F(y,t)dV(y)$$

Then every local fluid parameter F may be considered as an average plus a space deviation:

$$F \cong < F >_f + \delta F$$

Let $\chi_k(x,t)$ be the phase characteristic function for phase k or field characteristic function for field k (k=1, n)

$\chi_k(x,t)$ = 1 when point x is in the phase k or field k at time t

$\chi_k(x,t)$ = 0 when point x is not in the phase k or field k at time t

One can also multiply by the product $\chi_k(x,t).\chi_f(x)$ for a two-fluid or multi-field model in a porous body approach.

After averaging of the basic equations multiplied by $\chi_f(x)$ or $\chi_k(x,t).\chi_f(x)$ for k = 1, n, the three balance equations (mass, momentum and energy) are written n times one for each phase or field.

Here below are the equations written for a two-fluid model after the time averaging.

$$\frac{\partial \alpha_k \overline{\rho_k}^k}{\partial t} + \frac{\partial \alpha_k \overline{\rho_k v_{kj}}^k}{\partial x_j} = \Gamma_{ik}$$

$$\frac{\partial \alpha_k \overline{\rho_k v_{ki}}^k}{\partial t} + \frac{\partial \alpha_k \overline{\rho_k v_{ki} v_{kj}}^k}{\partial x_j} + \frac{\partial p}{\partial x_i} = \alpha_k \overline{\rho_k}^k g_i + \mathrm{F}_{ik} - \frac{\partial \alpha_k \left(\overline{\sigma_{ij}}^k + \overline{\rho_k v'_{kj} v'_{ki}}^k\right)}{\partial x_j}$$

$$\frac{\partial \alpha_k \overline{\rho_k u_k}^k}{\partial t} + \frac{\partial \alpha_k \overline{\rho_k v_{kj} h_k}^k}{\partial x_j} + \alpha_k \overline{v_{kj}\frac{\partial p_k}{\partial x_i}}^k = \alpha_k \overline{\rho_k v_{ki}}^k g_i + \mathrm{Q}_{ik} - \frac{\partial \alpha_k \left(\overline{\sigma_{ij} v_{kj}}^k\right)}{\partial x_j} + \frac{\partial \alpha_k \left(\overline{q_{kj}}^k + q_{kj}^t\right)}{\partial x_j}$$

The e two-fluid approach or a (3 x n) equation model for an n-field model.

Space averaging of these equations is also applied to obtain the final system of equations. After averaging of local instantaneous equations, many terms of the resulting equations are new unknowns or contain new unknowns. At this step, there are more unknowns than equations and the system of equations is not closed and cannot be solved. In particular, averaged equations exhibit terms for transfers of mass, momentum and energy at the walls and at the interfaces (in two-phase or multi-fluid flow conditions). Constitutive relations are expressions for these transfers as functions of the selected averaged principal variables. Other closure relations are also necessary to express the average of non- linear terms as functions of averaged main variables. These are also called closure relations since they are necessary to close the system of equations. Some closure relations are simplifying assumptions such as:

se three balance equations are written for each phase or each field resulting in a 6-equation model for th

<P (P, T)> = ρ (<P>, <T>).

Let us see the 1D 2-fluid equation where a cross section averaging is applied for flows in pipes. The formulation below results from double time and space averaging and from many simplifying assumptions. All flow variables are here double-averaged quantities.

$$A\frac{\partial \alpha_k \rho_k}{\partial t} + \frac{\partial A \alpha_k \rho_k V_k}{\partial z} = A\Gamma_{ik}$$

$$A\frac{\partial \alpha_k \rho_k V_k}{\partial t} + \frac{\partial A \alpha_k \rho_k V_k V_k}{\partial z} + A\alpha_k \frac{\partial P}{\partial z} = A\, \alpha_k \rho_k g_z - AF_{ik} - \chi_w \tau_{wk}$$

$$A\frac{\partial \alpha_k \rho_k \left(H_k + {V_k}^2/2\right)}{\partial t} + \frac{\partial A \alpha_k \rho_k U_k \left(H_k + {V_k}^2/2\right)}{\partial z} - A\alpha_k \frac{\partial P}{\partial t}$$
$$= A\, \alpha_k \rho_k g_z V_k \quad + \chi_w q_{wk} + AQ_{ik} + A\Gamma_{ik}(H_k + Vi^2/2) + AF_{ik} V_k + \chi_w \tau_{wk} V_k$$

Several simplifications were necessary to obtain this form of the equations. In particular:

– The single pressure assumption assumes that there are algebraic relations between the averaged gas pressure, the averaged liquid pressure, and the averaged interface pressure. Only one mixture pressure P appears in the set of equations but additional terms due to pressure differences between phases and interface may exist in interfacial momentum transfers and in wall friction terms where singular pressure drops may be added to the regular friction.
– Many averages of products are assumed equal to the products of averages.
– State equations are assumed valid for averaged variables.
– Axial turbulent transfers are neglected.

The successive steps to establish the model equations and to solve them are shown in Fig. A-4.1. The space discretization of space and time averaged equations for all modules (0-D, 1-D, 2-D and 3-D) the results in a set of ordinary differential equations and further time discretization, leads to a set of linear algebraic equations.

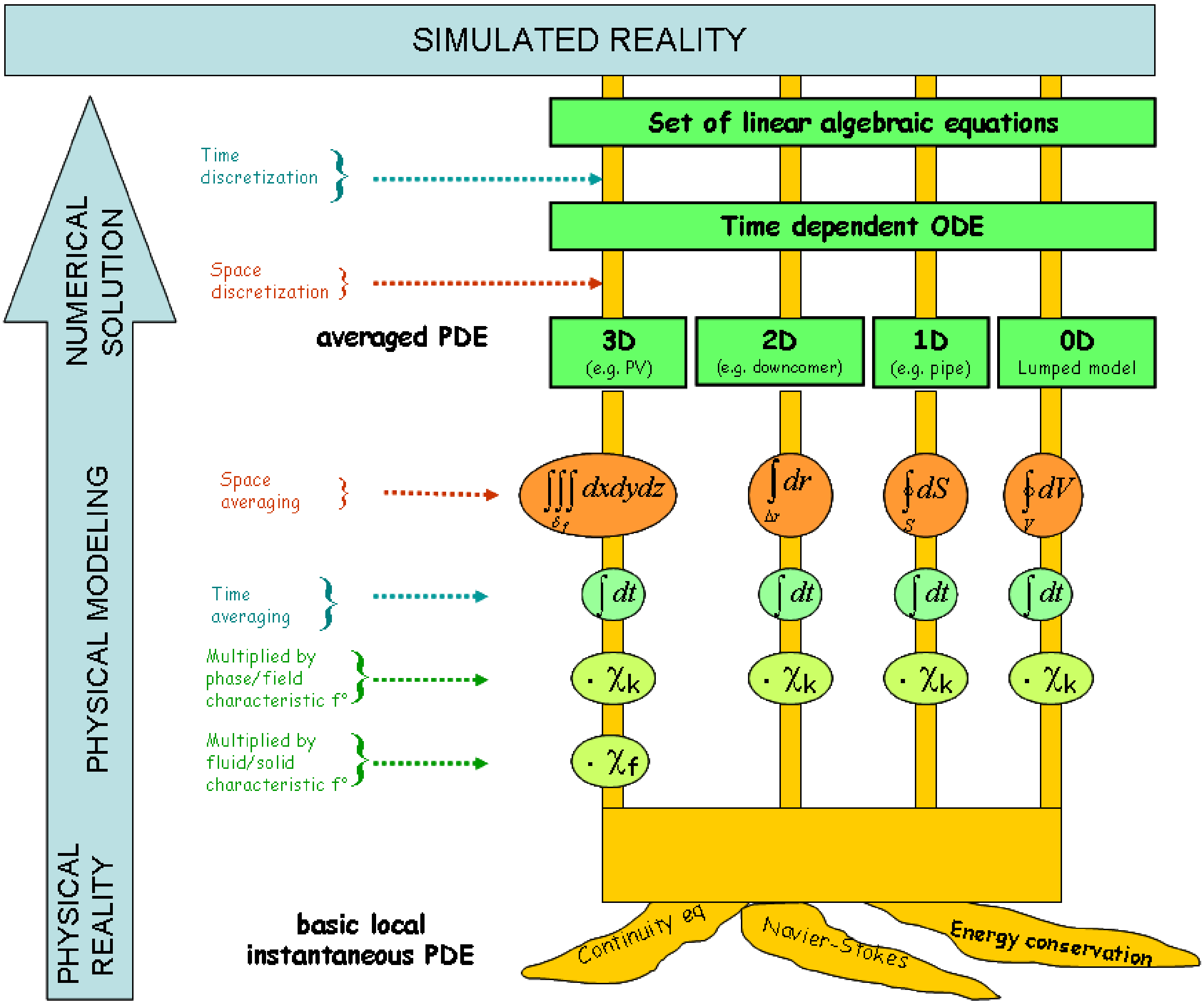

Fig. A-4-1 – The successive steps for establishing and solving thermal-hydraulic equations.

On the effect of averaging on scaling capabilities

There is no doubt that the averaged equations used in system codes have the scaling capabilities corresponding to the top-down approach of the H2TS or to the system level analysis in a FSA. It means that they can actually identify easily the major contributors to a mass inventory or a system pressure evolution in a transient. The averaging may become a problem for phenomena identified in the bottom-up approach of a H2TS or on component or process levels of a FSA. For such phenomena, attention should be paid to the time and space resolution, and to the formulation of closure laws. The averaging process is intrinsic property of the system codes and an effect of averaging or process model on the figure of merit is expected.

A-4.1.2 The constitutive relations and/or closure relations

In the 1-D two-fluid 6-equation model above one may identify 14 (fourteen) transfer terms which require constitutive relations:

- Wall transfers
  - Liquid-wall friction
  - Gas-wall friction
  - Wall-to-liquid heat transfer
  - Wall-to-gas heat transfer
- Interfacial transfers
  - Interfacial mass transfer (vaporization or condensation)
  - Liquid to interface heat flux
  - Energy transfer to liquid due to interfacial mass transfer
  - Gas to interface heat flux
  - Energy transfer to gas due to interfacial mass transfer
  - Momentum interfacial transfers:
    - Interfacial friction,
    - Virtual mass force (or added mass force),
    - Differential terms of momentum equations related to pressure differences between phases and interfaces,
    - Momentum transfer to liquid due to interfacial mass transfer,
    - Momentum transfer to gas due to interfacial mass transfer.

<u>Flow regime maps in SYS TH codes</u>

Two phase flow may exist under various flow regimes characterized by specific phase distributions. Every flow regime has its internal structure and its transfer mechanisms. So it seems natural to use a flow regime map (or flow pattern map) in a code and to develop constitutive relations for wall and interfacial transfers which depend on the flow regime.

<u>Geometry of the flow in NPPs</u>

The flow regimes and the structure of phase distribution may depend also on the flow geometry. In NPPs one may encounter various flow geometries:

- horizontal or vertical tubes (0.01 < Dh < 1.3 m)
- rod bundles (core geometry) or tube bundles (steam generators)
- annuli (downcomer geometry)
- large Volume (Pressurizer, Upper Head, headers, steam dome of a steam generator)
- large Volume with internal structures (Upper plenum, Lower plenum)
- (singular geometries: bends, flow restrictions, nozzles, separators, dryers…)

As a consequence system codes must include constitutive relations which depend on the flow regime, and on the geometry.

Phenomenological and empirical constitutive relations

A purely empirical correlation is a best fit of experimental data where the quantity to model is expressed as any function of the principal variables. It can be very accurate within the domain of experimental investigation but the extrapolation beyond it is very dangerous. On the other hand, this method does not take any benefit from the knowledge which may exist in certain sub domains where physically based models are available.

Dimensional analysis allows in principle to determine the dimensionless numbers to use in the expression of the quantity to correlate. But in 2-phase conditions, the number of independent parameters is very high so that simplifying assumptions are necessary. When the controlling physical processes are well identified, one can keep only the few dimensionless parameters which play a role. In this case, the extrapolation beyond the investigated domain is less hazardous. Nevertheless there is no guarantee since the controlling processes can be different in another range of parameters.

The phenomenological or mechanistic approach consists in assuming a governing physical mechanism. The correlation is then derived theoretically without any input from experiments. An alternative is to keep some free parameters to adjust based on experimental data. This semi-empirical approach was frequently used in the current system codes. Even with this last precaution, the extrapolation beyond the qualified domain is not guaranteed. New effects, which are not present in the model, may become important in another range of parameters. The experience showed that 2-phase thermal-hydraulics contains myriads of phenomena which make it difficult to generalize any theoretical model. Interpolation is often possible but extrapolation is dangerous even with physically based models.

<u>Well-posedness of the system of equations</u>

The system of equations such as the 6-equation here above may be written in the following form:

$$A \,{}^{\partial U}\!/_{\partial t} + B \,{}^{\partial U}\!/_{\partial z} = C$$

When the characteristic equation has six real roots (characteristic velocities) the system is hyperbolic:

$$Det(B - \lambda A) = 0$$

The characteristic velocities of the 1D 2-fluid 6-equation model are:

- v_g transport velocity of gas enthalpy h_g
- v_l transport velocity of liquid enthalpy h_l
- $w_s - c_s$, $w_s + c_s$ pressure wave propagation velocities
- $w_\alpha - c_\alpha$, $w_\alpha + c_\alpha$ void wave propagation velocities

The single pressure two fluid convective equations are not hyperbolic without differential forces (added mass force, force due to pressure differences between phases and interface and proportional to the void fraction gradient, 'grad-α').

Hyperbolicity is a condition of the well-posedness of the problem and a condition for the stability. However non hyperbolic equations are used in some system codes and remain stable through diffusion terms or numerical diffusion. If the stability only depends on numerical diffusion, instability may occur when the mesh size is small enough to reduce the stabilizing effects of numerical diffusion. Therefore codes which use ill-posed equations cannot do mesh convergence tests with very small mesh size.

Even when adopting hyperbolic equations, the presence of junctions between modules which give mesh constraints does not allow achieving mesh convergence for the modelling of reactors.

A-4.1.3 Lumped model or 0-D model

For the simple calculation, some components in an NPP may be modeled in a lumped parameter way, i.e. a lumped model. A lumped model solves the whole volume of the component by single set of balance equations. Mass and energy conservation equations are averaged over the component volume. They are Ordinary Differential Equations (ODE) whereas Partial Differential Equations (PDE) are written in 1-D or

3-D models. No internal velocity field is calculated in the component; only flow-rates are calculated at boundaries of the component or at the junctions between the component and adjacent components. Momentum balance equations are written at each junction to calculate flow-rates as function of pressures and other scalar quantities on both sides of the junction. A lumped model is a 0-D model.

However, assembling several adjacent lumped models or Control Volumes for modelling a pipe may result in a system of equations which resembles the 1D model or even 3-D model given here above. The analogy is even stronger when equations are discretized with the staggered grid method with scalar variables defined at cell centers and with vector variables defined at cell faces.

A-4.1.4 Physical model in 2-D or 3-D models

Constitutive relations used in 2-D and 3-D models are generally extrapolated or simply taken from 1-D models. This may lead to some shortcomings. In particular, wall friction tensor and interfacial friction tensors are idealized and may be simplified to isotropic or even scalar. The main problem is associated with the lack of turbulent diffusion modelling in present 2-D or 3-D models implemented in system codes. These models should be used only when the turbulent diffusion effects are dominated by other effects.

A first example is the core, a very porous medium where the diffusion towards rod walls or interfaces is much higher than the large scale turbulent diffusion. Moreover, in low velocity two phase conditions, gravity effects are likely to produce the most important large scale mixing effects. The lack of turbulent diffusion terms is not restrictive in this case.

A-4.1.5 Special thermal-hydraulic models

Thermal-hydraulics of coolant fluid in reactor circuits may be influenced by a local specific geometry, which requires some specific models. Some of them are classified as special component models or special process models. Examples of special process models are:

<u>Critical flow</u>

Choked flow or critical flow conditions may occur in reactor transients either at a break or in internal flow at flow restrictions or pumps. When sonic velocities are reached at a section (often at the smallest flow cross section area) the flow becomes independent from downstream conditions. Predicting critical flow is then of prime importance for all LOCA transients since the break flow rate controls the coolant mass inventory and distribution, and consequently the core cooling capability.

System codes developed either 0-D or 1-D models for break flow or chocked flow but in the worst conditions the uncertainty of prediction can reach 20% or more due to complex geometrical effects and to unpredictable nucleation conditions. Assessment of TRAC code with Marviken as part of CSAU uncertainty estimate showed that critical flow uncertainty was in excess of 20% [CSAU, NUREG-CR-5249], USNRC, 1989.

<u>Singular pressure losses</u>

Complex geometries such as bends, flow limiters, valves, sudden area change, nozzles, perforated plates, spacer grids, support plates, induce pressure losses which are called “form losses” or “singular pressure losses”. System codes do not predict these losses and the user has to enter loss coefficients in the input deck.

<u>Counter-Current Flow Limitation (CCFL)</u>

The flooding limit and the Counter-Current Flow Limitation plays an important role in many accident sequences since it may control the quantity of cooling water which is kept out of the core (Upper plenum, Hot Leg, SG tubes) and which is no more available for the core cooling. The basic phenomenon is the limitation of a liquid downward flow-rate for a given upward gas flow-rate. The CCFL is likely to occur in complex geometry such as Upper Core Tie Plate, Hot Leg bend, inlet of Steam Generator header, inlet of Steam Generator Tube, and a specific local modelling is necessary in system codes to obtain reliable predictions. Flooding correlations are established from experimental data and the parameters of these correlations are entered in the code input deck; see e.g. Kinoshita et al., 2012.

Examples of special component models are:

Separators

Separators are used to separate steam and liquid water at core exit of BWRs or at the top of PWR Steam Generators. A two-phase mixture enters the separator barrel, passing through a set of stationary swirl vanes. These vanes produce a high rotational velocity component in the fluid flowing through the separator barrel. The resultant centrifugal force separates the steam-water mixture into a water vortex on the inner wall of the separator barrel and a steam vortex core. Because this process cannot be simulated by using the 1-D two-fluid model, special models are used for separators. The separator model determines the void fraction in the (bottom) liquid fall back junction and the liquid fraction in the (top) outlet junction. In general, two options might be available; (i) a simple separator model in which a steam-water inflowing mixture is separated by defining the quality of the outflow streams using empirical functions, and (ii) a mechanistic separator model, which is intended to model the centrifugal separators by solving phasic continuity equation, angular momentum equation, and axial momentum equation with some simplifying assumptions.

Dryers

The steam dryer uses Chevron vanes to remove the moisture which is discharged from the steam separators. The vanes provide a curved path which the liquid droplets must follow if they are to flow through the dryer. The liquid droplets, flowing along the curved path, hit the vanes due to their inertia and are de-entrained. The resultant liquid film flows down the vanes under the force of gravity and then back to the liquid pool surrounding the separators. The dryer efficiency depends on the steam velocity and the moisture content of the steam flow entering the dryer. Since this process cannot be simulated by using a 1-D two-fluid model, special models which are either simple or mechanistic are used for dryers.

Pumps

Pumps are generally modeled as a point (or 0-D) module. A pump is located in a node or a mesh and it adds source and/or sink terms to the momentum and energy balance equations. The angular momentum equation is added and controls the rotation speed of the pump. The hydraulic torque is generally modeled through user defined characteristic functions of fluid velocities and rotation speed. The hydraulic torque is a sink term for the angular momentum equation and is multiplied by the rotation speed in a source term for the fluid energy equations. The pump Head is also modeled through user defined characteristic functions of fluid velocities and rotation speed. The pump head appears as source terms in the fluid momentum equations. The head and torque characteristic functions have to be known in the four quadrants.

Usually pump vendors give only single phase characteristic functions in the first quadrant. It may be necessary to do experimental tests to get the pump characteristic functions in the other quadrants. Moreover two-phase characteristic functions have also to be given as user defined functions since pump performance (head and torque) are degraded in two-phase flow conditions. Usually two-phase characteristic functions are also function of inlet void fraction.

Turbines

Turbines, like the pumps (see above), are generally modelled as a point (or 0-D) module. Turbine also adds source and/or sink terms to the momentum and energy balance equations. The sink and source terms are also modelled as user defined functions.

ECC injections

ECC injection is often simply modelled as a source term (of mass, momentum and energy) in a node or mesh. A very simple model may be used when the flow-rate is a function of the pressure at injection location. The ECC injection (special) model takes care of the mixing, including interfacial transfer, between different fluids in the delivery line and in the primary system (typically).

Accumulators

Accumulators may be modelled either using standard modules (0-D, 1-D, valves, Tee, walls) of the codes or by a specific accumulator model. The specific module must predict:

- Accumulator pressure as function of remaining mass and of heat released from accumulator walls (from isentropic depressurization for rapid discharge to quasi-isothermal for slow discharge).
- Flow-rate delivered to the circuit as function of valve opening, accumulator pressure, primary circuit pressure, pressure losses, etc.
- Mass momentum and energy source terms in both phases in the injection mesh.
- If some specific options are actuated, the accumulator specific module should also predict the amount of dissolved nitrogen in liquid water and the source term of nitrogen in gas mixture in case of transport of non-condensable gas.

Valves, safety valves, control valves, check valves and flow limiters

Many valves, safety valves, control valves, check valves exist in a reactor. They may be “internal” to the modelled circuit or “external” when they connect the modelled circuit with the external space (e.g. containment or pressure suppression pool). They are characterized by functions giving either the pressure loss as a function of flow-rate and the degree of opening, or flow-rate as function of upstream pressure and temperature for choked flow conditions. Most of the time, only single phase valve characteristics are known. The valve model also must predict the behaviour in abnormal two-phase conditions.

Breaks

The break may occur anywhere in a pressurized system and may put in communication a high pressure (typically high temperature) environment with a low pressure (typically ambient temperature and pressure) environment. Conditions for the occurrence of ‘critical flow’ (i.e. Two-Phase Critical Flow, TPCF) develop at the break. Although a number of breaks and break configurations are excluded for technological reasons (e.g. catastrophic vessel break, application of the Leak before Break concept in selected piping), a wide variety of break configurations may appear. This reflects in a wide variety of break shapes that cannot be controlled by thermal-hydraulic parameters and, at the same time, affects the prediction of relevant thermal-hydraulic quantities like critical flow-rates. Furthermore, the capabilities to predict TPCF are not necessarily embedded into the balance equations (with the noticeable exception of the CATHARE code, where a specific experimental database was created and used) even due to the specific ranges of validity of relevant equations: e.g. friction pressure drop at very high fluid speed, or vaporization rate when steep pressure gradient occur along the flow direction and as a function of time. Owing to this, special critical flow models are needed and are provided by code developers as well as by several researchers in the open literature.

Spray cooling

The spray of sub-cooled liquid into a steam environment constitutes an efficient way to cool a space-region of the nuclear power plant. Spray cooling is adopted in nominal operating conditions (e.g. cooling of the vapour region of the pressurizer in PWR in order to control the pressure) or in accident conditions (e.g. cooling of the upper core region of BWR to facilitate the liquid entering into the upper part of the core following the occurrence of accidents). Thermodynamic non-equilibrium phenomena occurring in direct-contact-condensation conditions are involved, that are not within the domain of validity of closure equations. Furthermore, the phenomenon is controlled by aspects like the droplet size, the droplet speed distribution, and the formation of droplets clouds and the interaction of droplets with solid walls. Those aspects are not part of the balance equations; rather they are dependent upon the design and the operation of the spray nozzles. Owing to this, special models are needed and are provided by code developers as well as by several researchers in the open literature, e.g. see Dix & Andersen, 1978, for more details.

Physical models for non-thermal-hydraulic systems

Thermal-hydraulics evolution of coolant fluid in an NPP is affected by phenomena, components and systems which are not strictly thermal-hydraulic phenomena components or systems. Examples are neutrons generated fission power, I & C, hydrogen production. The physical performance of those phenomena, components and systems has to be calculated and the influence on thermalhydraulics has to be modelled.

A-4.2: Code verification

A-4.2.1 the various steps of code verification

Verification is a process to assess the code correctness and numerical accuracy of the solution to a given physical model defined by a set of equations. In other words, verification is performed to show whether the equations are correctly solved by the code. Thus, the relationship of the calculation results to real world is not an issue in verification of the code. Simply speaking, verification deals with mathematics and data processing. In a broad sense, the verification is performed to demonstrate that the design of the code numerical algorithms conforms to the design requirements, that its logic is consistent with the design specification, and that the source code conforms to programming standards and language standards.

Verification activities start in the context of the code development process. Let's assume that physical phenomena and processes to be simulated were represented by a set of equations and closure relations and that the requirements for the numerical solutions were determined. Then, the next activities are:

(a) The equation set are solved by using numerical solution method. That is, we establish a numerical algorithm and prepare the documentation in detail.
(b) Next, the numerical algorithm is implemented into a computer program, resulting in the source program written in a computer language and computing environment.
(c) The solutions of the code are assessed by comparing with highly accurate solutions.

In steps (a) ~ (c), verification activities are needed to demonstrate the correctness of the numerical algorithm, the source code in terms of software quality engineering (SQE), and the accuracy of numerical solution, respectively. The schematic for the verification process of a system code is shown in Fig. 4-2. The first two activities are called code verification and the last is called solution verification or calculation verification. The final result of the verification might be rather simple, i.e. satisfactory or not. In particular, we should separate the current best practice, mostly based on engineering judgment, from an 'ideal' practice; in this connection technological acceptance criteria could be introduced for each step of V & V.

The plan for code verification should be prepared early during the development of the code. This should preferably be done when the functional requirements of the code are being written. The plan should include objectives, approach, schedule, and plan for testing that includes a verification matrix, i.e. a set of test problems for verification with highly accurate solutions. The plan should be reviewed and updated as necessary.

Verification tasks should be assigned to the code developers. An independent verification process may be desirable. The results of all verification activities should be documented and preserved as a part of the system for software quality management.

It is noted that the verification should be carried out before the validation. If an error in the numerical algorithm is found in the validation step, the error should be fixed and its effect should be assessed in terms of verification before proceeding with the validation.

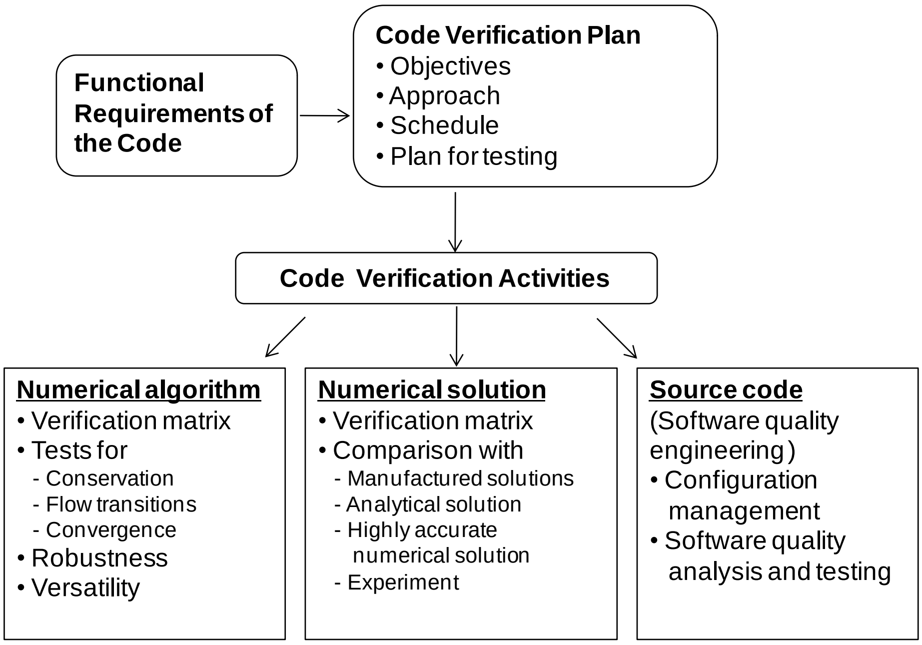

Fig. A-4-2 –Diagram for verification of a system code.

Numerical algorithm and numerical solution

The verification of numerical algorithm and solution may be simultaneously carried out. However, because of the complicated nature of physical phenomena and processes to be modeled in a system code, the separate verification approach is more efficient and, thus, desirable. The objectives of the numerical algorithm verification is establishing confidence that the numerical algorithm works correctly in accordance with functional requirements through removing programming and logic errors in the computer program. This step begins with a review of the documents involving the design concept, basic logic, flow diagrams, numerical methods, algorithms and computational environment. Emphasis is put on qualitative analyses under various possible situations. In other words, the operability (i.e. the capability to deal with different flow conditions) of the numerical scheme, is a key target for the numerical algorithm verification. Comparisons with 'reference' highly accurate solutions may not be important and, thus, we may have greater flexibility to select test problems. When this step is successfully completed, versatility and robustness of the code can be established.

The target of the numerical solution verification is to confirm that the numerical equations are correctly set up and the solution is correct. The effective way to confirm the accuracy is to compare with known test problem having highly accurate solutions. Therefore, accuracy definition and numerical error estimation are important issues in this step. The errors that should be assessed in the solution verification step are purely numerical, which are mainly due to discretization and round-off. These errors are relatively small in comparison with the errors from the physical models, and can be quantitatively estimated by using numerical tests, such as a mesh convergence test when they are possible. If the numerical solution error exceeds a certain value, the cause should be identified and fixed.

Numerical algorithm

The numerical algorithm is the set of mathematical algorithms which is used to solve the set of non-linear PDEs. It includes the space and time discretization of equations which is used in Finite-Volume or Finite-Difference methods. Then the set of non-linear PDEs is finally converted to a set of coupled linear equations. The converged solutions in one time step are calculated using iteration technique in most cases. Many requirements are associated to the numerical algorithm. Examples of common requirements are given below.

Some requirements for the numerical algorithm are **mandatory** such as th*e consistency,* the *convergence* and the *stability*. Some other requirements are **very important** for system codes:

1. *Robustness*: capability to converge to a solution in all physical situations of the domain of simulation without code failure. If the system code is used to design nuclear power plants or to assess the safety of them, this requirement is very important.
2. *Mass and energy conservation*: It is not only a requirement for being accurate on mass and energy but also to be able to predict system pressure correctly since any numerical source or sink of mass or energy will affect the pressure. The system pressure has a strong impact on reactor transient scenarios since it controls actuations of Scram signal, Safety Injections, Safety valve opening, Accumulator discharge, and so on. Mass conservation affects only thermal-hydraulics, whereas energy conservation affects also heat structures.

Some requirements are also **important** for system codes:

- *Numerical efficiency criterion:* The CPU time must remain compatible with the intended use of the code.
- *Accuracy*:
  - Accuracy on mass and energy conservation is mandatory (see above)
  - Accuracy of transport processes has to be sufficient with respect to some acceptance criteria, or at least requirements on space and time increments (dz & dt) may be defined to obtain a given accuracy. This may be important for transport of a temperature front (e.g. positive reactivity feedback due to flowing cold water into the reactor core by overcooling in the steam generator in case of a steam line break) or for the transport of a boron concentration (e.g. positive reactivity feedback due to flowing un-borated water into the reactor core in case of small break loss of coolant accident).
  - Accuracy of propagation processes has to be sufficient with respect to some acceptance criteria, or at least requirements on space and time increments (dz & dt) may be defined to obtain a given accuracy. This may be important for pressure wave propagation in water hammer situation or in condensation-induced water-hammer where strong pressure peaks may propagate in the fluid. This may also be important for void fraction wave propagation in

particular in case of stratified flow where free surface waves may propagate either downstream or upstream depending on the conditions. In subcritical conditions, the prediction of void fraction requires the capability to predict the upstream propagation velocity correctly.

- Accuracy and numerical efficiency are tightly coupled since any numerical scheme may improve its accuracy by reducing space and time increments (dt & dx), which also increases the resolution time.

- The numerical algorithm has to be able to deal with water level, swollen levels, and water-packing. When a liquid level of swell level passes a mesh there should not be numerically induced pressure peaks that affect the transient behavior.

At last, there are several requirements related to operability (i.e. the capability to deal with different flow conditions) of the numerical algorithm to treat the physical situations of interest:

- The numerical algorithm has to be able to deal with incompressible and compressible flows with non-simplified equations of state
- The numerical algorithm has to be able to deal with flows with strong interfacial mass transfers (strong pressure-void fraction coupling)
- The numerical algorithm has to be able to deal with strong hydraulic-wall conduction couplings (passing DNB, rewetting)
- The numerical algorithm has to be able to deal with flows in an extended range of velocities including subsonic and supersonic velocities
- The numerical algorithm used for thermal-hydraulics equations has to be able to deal with all the two-phase flow regimes in the whole range of void fraction and to respect some positivity constraint to keep the void fraction in the range [0, 1] while preserving mass and energy conservation.
- The numerical algorithm has to be able to work in a wide range of pressures and temperatures: $0 < Tl < 350°C$; $0 < Tv < 2000°C$; $0.001 MPa < P < 22 MPa$
- When used with additional equations for non-condensable gases, the numerical algorithm must respect some positivity constraint to keep all gas mass fractions (or molar fractions) of each component of the gas mixture (e.g. steam-air, steam-H_2, steam-N_2) in the range [0, 1] while preserving mass and energy conservation.

Verification matrix for numerical algorithm and solution

Numerical scheme is verified by comparing the results of the code calculations with highly accurate solutions. For systematic verification, a set of highly accurate solutions, called a verification matrix, should be established so that various possible solutions of the numerical scheme can be compared with them, finally being proven to work as intended. Of course, the verification matrix must be able to check whether the numerical scheme actually meets the numerical requirements listed above.

When establishing a verification matrix, a systematic selection of the test problems is very important. For example, in the verification of the two-fluid equations, we have to consider test problems with various flow conditions, including single-phase liquid flows, two-phase flows, and single-phase vapor flows. The possible flow transitions also should be included in the test problems. In addition, a two-phase flow has various flow regimes and, according to the flow regimes, there are various constitutive models, such as wall friction, wall heat transfer, interfacial friction, and interfacial heat transfer, etc. The test problems should be selected so that all these models can be verified. Considering complicated physical models and various possible flow conditions, we should establish an extensive verification matrix. The algorithm verification using simple conceptual problems is very important, especially in the early stage of a code development.

Each problem in the verification matrix should be documented, containing four elements; (i) conceptual description, such as general classes of physical processes, initial and boundary conditions, and numerical algorithms that are being tested, (ii) mathematical description to provide an unambiguous, reproducible mathematical characterization of the benchmark problem, (iii) accuracy assessment of the solution, and (iv) additional user information.

A verification matrix can be used in common for both the numerical algorithm verification and solution verification. Because the former is rather focused on qualitative comparisons with various cases, comparisons with highly accurate solutions are not much important as mentioned previously and, thus, we may have greater flexibility to select test problems. However, since the latter deals with the numerical accuracy of the code, problems having highly accurate solutions should be selected.

Four types of solutions are available for verification of the numerical scheme. These include manufactured solutions, analytical solutions, highly accurate numerical solutions of ODEs and PDEs or benchmark, and experimental benchmark.

1. Manufactured solutions

Manufactured solutions are specifically constructed for testing numerical algorithms and computer codes, (see Oberkampf and Trucano, 2008, and Roache, 1998). The method of manufactured solutions allows one to custom-design verification solutions by altering the original PDEs of interest in the mathematical model. A specific form of the solution function is chosen and then the original PDE of interest is modified such that the chosen solution function satisfies the modified PDE. The solution function is inserted into the original PDE, and all the derivatives are obtained through symbolic manipulation. The equation is rearranged such that all remaining terms in excess of the terms in the original PDE are grouped into a forcing-function, or source term, on the right-hand side of the PDE. With this new source term, the assumed solution function satisfies the new PDE exactly. When this source term is added to the original PDE, one recognizes that we are no longer dealing with physically meaningful phenomena, although we remain in the domain of mathematical interest. Note that this method mainly verifies the differential terms on the left hand side of the equations since algebraic source terms on the right hand side are modified to be consistent with the predefined manufactured solution.

2. Analytical solutions

Analytical solutions are closed-form solutions to special cases of the PDEs defined in the mathematical model. The most significant practical shortcoming of classical analytical solutions is that they exist only for very simplified physics, material properties, and geometries. However, with some assumptions for simplification, analytical solutions to one-dimensional two-phase flows can be obtained, e.g. a transient solution for manometer flow oscillations in a U-shape tube is easily obtained. The solution for a phase separation by gravity is also available. In these cases, the source program should be modified to be consistent with the assumptions that were used to obtain the analytical solutions. In spite of some source program changes, these are still useful for the verification of the source program.

There are more examples of analytical solutions as conceptual problems in the verification.

(a) Consider a steady-state, one-dimensional, single-phase flow in a horizontal pipe. If the inlet flow rate is zero and the exit pressure is a constant, then the pressure and velocity in the pipe are clearly known. By increasing the inlet flow, only the pressure drop increases. The calculation results can be compared with nearly exact solutions. By adding volumetric heat source in the fluid, the fluid temperature increases, which are known with known mass flow rate and specific heat. By using these types of nearly exact solutions, basic verification tests for mass, momentum, and energy conservation can be conducted. Mesh convergence can be checked as well.

(b) A closed one-dimensional horizontal loop, which is filled with water at a constant temperature, also provides a nearly exact solution. If the water velocity is given, it experiences an exponential decay with the assumption of a constant friction coefficient. If the flow is initially stagnant in the loop, the flow should continue to be stagnant. This can be used to verify the momentum equations.

As shown in above examples, each analytic solution is made for the verification of specific equation set in a system code. It is noted that, when generating a simplified analytical solution, the problem should be designed to aim at the verification of specific terms in the governing equations.

3. Highly accurate numerical solutions of ODEs and PDEs and benchmark solutions

Highly accurate solutions consist of numerical solutions to special cases of the general PDEs that can be mathematically simplified to ODEs or numerical solutions to more complex PDEs. The accuracy of numerical solutions to more complex PDEs clearly becomes more questionable when such solutions are compared with manufactured solutions, analytical solutions, or ODE solutions. Computational results with well-qualified codes can be considered to be "benchmark solutions" for verification. But the reliability of these solutions is itself a factor that is hard to separate from the verification task. Thus, the use of these solutions for SYS TH code verification is very limited in practice.

4. Experimental benchmark

Well-defined experimental results also can be used for verification of the numerical scheme. But a strict comparison of the computational results with the experiment is of limited value here because of measurement errors. Thus, a qualitative comparison is desirable for verification and a quantitative comparison can be carried out later in the context of validation.

A-4.3: Code validation

Validation is the process to assess the adequacy of the physical models of the code. Physical models include some first principles laws which do not require any validation and many closure relations which are simplified descriptions of the flow processes and which require a validation. The main aspects of these physical laws and closure relations of current SYS TH codes are first summarized, focusing on thermal-hydraulic models. However, non-thermal-hydraulic models such as Neutron Kinetics, fuel thermo-mechanics, Hydrogen production also contain simplifications and closure relations to be validated. Then some characteristics of Validation matrices are given with the selection criteria, the role of different kinds of tests, and the way code results are analysed. The content of validation report is defined. The role of validation in code uncertainty methods is presented. The role of sensitivity tests during the validation process is explained. The development and qualification of nodalization is addressed and the relations between validation, User Effect and User Guidelines are discussed.

A-4.3.1 Physical laws and closure relations

System codes model the physical system - the reactor cooling circuits, or test facility - to be simulated by writing systems of equations. The equations express the basic principles of physics such as mass conservation, Newton law, first principle and second principle of the thermodynamics. These are physical laws which are universally valid. However, in fluids these physical laws are applied in system codes with some approximations related to space and time averaging. This averaging allows to focus on some macroscopic phenomena of interest and to avoid spending an unaffordable CPU time in solving all the microscopic details of the flows. Since these equations simplify the physical reality, they are no more exact equations, they are called models. Due to the averaging of local instantaneous basic equations, many additional terms appear in the balance equations which require closure relations or constitutive relations.

Such averaged equations are no more exact equations but rather equations which model the physical reality, as close as possible. This is the reason why Validation is necessary since one must measure or estimate to which degree these equation approximate the reality.

In the 1-D two-fluid 6-equation model which is used in most current system codes, one may identify fourteen transfer terms; see the list in section A-4.1, which require constitutive relations.

All those transfer terms may depend on the flow regime. One distinguishes flow regimes in pre-CHF or post-CHF condition in either horizontal or vertical flow.

Flow regimes in pre-CHF conditions are:

- Bubbly flow
- Cap bubble flow
- Slug – churn flow
- Annular flow
- Annular – mist flow
- Stratified smooth
- Stratified wavy
- Stratified mist flow
- Plug-slug flow
- Dispersed droplet flow

Flow regimes in post-CHF conditions are

- Inverted annular flow
- Inverted slug flow
- Dispersed droplet flow

Wall heat transfer regimes also depend on flow regime and on pre-CHF, transition, or post-CHF condition:

- Laminar natural convection to liquid
- Turbulent natural convection to liquid
- Laminar natural convection to gas
- Turbulent natural convection to gas
- Laminar forced Convection to liquid
- Turbulent forced Convection to liquid
- Laminar forced Convection to gas
- Turbulent forced Convection to gas
- Sub-cooled boiling
- Saturated boiling
- Critical heat flux
- Transition boiling
- Inverse annular film boiling
- Inverse-slug film boiling
- Dispersed flow film boiling

– Laminar wavy film condensation
– Turbulent wavy film condensation
– Radiation to inverse annular flow
– Radiation to inverse slug flow
– Radiation to dispersed droplet flow

Radiation modelling is rather simple in system codes. It may model only wall-to-fluid radiation in case of wall superheating (during core uncovery or reflooding situations). It may also take into account rod-to-rod radiation heat transfers or rod-to-housing heat transfers for rod bundles in accidental situation. Radiation modelling (e.g. emissivity) constitutes an area for improvements in the future.

Turbulent diffusion of momentum and energy is particularly important in the direction transverse to the flow and has a minor role in the axial flow direction. Therefore axial turbulent diffusion is generally not directly modeled in 1-D models of system codes. In such 1-D models, the transverse turbulent diffusion is modeled in the wall-to-fluid momentum and heat transfers. 3-D modules of system codes may have some turbulent diffusion modelling in all three directions or may neglect it. In most typical two-phase conditions, interfacial and wall transfers are about an order of magnitude higher than the turbulent diffusion. However, this should be proved by quantitative evaluation to clearly identify when and where turbulent diffusion can actually be neglected.

The 6-equation model includes about 100 to 200 closure laws or correlations to express all the transfers in all flow regimes and heat transfer regimes.

Validation of the thermal-hydraulic model should be able to assess the validity of each of these closure laws separately and to prove that they are consistent with each other’s and to model correctly all flow processes encountered in code applications.

Most models or closure laws are established in steady (or quasi-steady) and established (or quasi-established) flow conditions. An established flow is a flow where phase velocities and void fraction are constant along axial direction or have slow variations. In reactor applications the models are used also in transient (non-steady) and non-established (or non-fully developed) flow conditions where they may not be justified. This may induce some quantitative errors that can be observed during the validation process which must also contain transient and non-established flow data. This may have implication in code application for scaling.

Specific components having a particular geometry require some specific thermal-hydraulic models which also have closure laws which approximate the physical behavior and which need to be validated. In particular models for critical flow, Counter-Current Flow Limitation, Separators, Dryers have to be validated.

As already mentioned, system thermal-hydraulic codes also model other systems and components. Examples are fuel rods, heat exchanging walls, neutron heat source, pumps, turbines, ECC injections, accumulators, breaks, heaters, spray cooling, valves, safety valves, control valves, check valves, heat exchangers between components or between circuits, etc. Many of those phenomena, systems and components also include physical models which approximate the physical behavior and which need to be validated.

A validation matrix usually includes four types of tests:

1. Basic tests (or fundamental tests)
2. Separate Effect Tests (SET)
3. Integral effect tests
4. Plant data

OECD CSNI collected a Code Validation Matrix of Separate Effects Test Data, NEA/CSNI 1993, from the international community and produced a NEA Data Bank.

A-4.3.2 Validation on Basic Tests

The "basic tests", sometimes referred to as fundamental tests are used to validate the thermal-hydraulic modelling on an idealized level.

These problems usually involve very simple geometries and may have an analytical solution. They do provide information on generic model capabilities and numerics used by the code. In this respect they may have both Validation and Verification aspects. For example, a basic test may check the capability to respect the Bernoulli equation in flow restriction or enlargement. This validates the formulation of momentum equation and verifies the adequacy of the discretized equation. The fundamental tests sometimes help to identify discontinuities between correlation ranges and at flow regime transitions.

Examples of basic tests are:

- Bernoulli test: to check the capability to respect the Bernoulli equation.
- Drain - Fill Problem: to check the ability to predict the motion of a water level across node boundaries without diffusing the void fraction gradient and to determine the accuracy in calculating the gravity head as the cells slowly drain and fill.
- Single Tube Flooding: The ability to predict flooding and counter-current flow in vertical pipes is examined by comparing predictions of CCFL to well-known correlations. The validation provides guidance on modelling CCFL for vertical pipes.

A-4.3.3 Validation on SETs

The separate effects tests are designed to validate some specific part of the whole physical model independently from the others. Separate effects tests may address:

- A specific basic flow process (flashing, critical flow, direct contact condensation, CCFL, lower plenum voiding, etc.)
- A specific closure term for a given flow or for a given range of flow parameters.
- The specific behaviour of a specific reactor component (separator, break, valve, loop seal, etc.). This may be called a component test.
- A specific flow process occurring in a specific reactor component during a selected time period of an accidental transient identified as phenomenological window (Reflooding, Refill)
- A specific closure term in a specific reactor component (or reactor zone).

Ideally a system code which must cover the DBA accidents should have a SET matrix able to cover:

- All closure terms for every flow regime and every heat transfer regime.
- All flow regime transitions and heat transfer regime boundaries (such as CHF, rewetting temperature, ONB, NVG, etc.).
- All flow geometries encountered in reactors (pipes, rod bundles, tube bundles, large volumes, annuli, etc.).
- All reactor components which may induce a specific behaviour.
- The whole domain of flow parameters (i.e. pressure (P) in the range 0.001 MPa to 22 MPa, steam temperature (Tv) up to 1200 °C, velocities from zero to supersonic velocities, void fraction from zero to 1, etc.
- The whole domain of geometric scale, i.e. up to reactor scale.

In practice it must at least be shown that the SET matrix covers the most important processes in the most important parameter range, for all components where important phenomena occur. Moreover scale effects should be investigated to allow at least a check of the scaling capability of the physical models.

The selection of separate effects tests should consider:

– Availability of well-defined initial and boundary conditions.
– Quality, quantity, and availability of the measurements.
– Well documented test reports and test results.
– Quality of measurement with estimation of uncertainty.
– Geometry representative of actual reactor.
– Parameter range.
– Scale as close as possible to reactor or scale complementary to other similar data at a different scale which will allow scaling capability of the physical models to be checked.

Validation on SETs is necessary:

– To check the validity of the physical models.
– To obtain the validity domain.
– To validate the scaling capability of the physical models.
– To estimate the uncertainties of the physical models.
– To define the best nodalization of each component.
– To define the best meshing and time step for converged solutions or optimal solutions.
– To improve current physical model.

The analysis of a SET calculation should contain:

– An identification of the specific effect to be validated and of the measured data to validate it.
– A justification of all choices made when creating the input deck including nodalization.
– A determination of the quantitative or qualitative accuracy of the prediction of the specific effect to be validated.
– An analysis of the reasons of the main code-experiments discrepancies using sensitivity tests on the nodalization, on the physical models or even on numerical options.

Although most models or closure laws are established in steady (or quasi-steady) and established (or quasi-established) flow conditions, the SET validation process must also contain transient and non-established flow data since in reactor applications the models are used also in transient (non- steady) and non-established flow conditions.

Due to the variety of models, the variety of flow regimes and the variety of geometrical configurations a SET validation matrix has a large number of tests. The NEA SET matrix, NEA/CSNI, 1993, has 2094 tests from 187 test facilities. The SET validation matrix of a recent CATHARE-2 version has about 1000 tests from about 50 test facilities.

The NEA SET matrix of experiments is suitable for the assessment of thermal-hydraulics transient system computer codes by selecting individual tests from selected facilities, relevant to each phenomenon. Correlation between SET facility and phenomena were calculated on the basis of suitability for model validation. The SET matrix is representative of the major part of the experimental work which has been carried out in the LWR-safety thermal hydraulics field, covering a large number of phenomena within a large range of useful parameters. The identification of phenomena for SET includes basic phenomena, critical flow, phase separation/vertical flow with and without mixture level, stratification in horizontal flow, phase separation at branches, entrainment/de-entrainment, liquid-vapor mixing with condensation,

condensation in stratified conditions, spray effects, countercurrent flow, heat transfer and global multidimensional fluid temperature void and flow distribution. In all, 67 hydraulic phenomena were used to relate to information provided from 187 test facilities used as potential sources of separate effects data.

Using these 187 SET facilities could be considered in 1990 as an optimum SET matrix for current LWRs at that time. Since then, new tests were performed which should be considered for validation (such as BFBT e.g. see Neykov et al., 2005, and PSBT, see Rubin et al., 2010).

A-4.3.4 Validation on IETs

IETs model the reactor under accidental conditions with all system components and all interactions between them. IETs are reduced scale tests. The IET matrix suitable for the validation of best estimate thermal-hydraulic computer codes consists of phenomenologically well-founded experiments, for which comparison of the measured and calculated parameters allows the evaluation of the accuracy of the code predictions.

The IET matrix should cover all transient types of interest. Examples of test types for PWRs are:

- Large Break LOCAs
- Small break LOCAs:
  - with HPIS
  - w/o HPIS
- Steam generator tube rupture (SGTR)
- Intermediate break
- Pressurizer leaks
- Loss of feed-water
- Loss of heat sink
- Station blackout,
- Steam line break
- Feed line break,
- Reactivity disturbance,
- Overcooling.
- Anticipated Transient Without Scram (ATWS)
- transients at shutdown conditions:
  - Loss of RHR at mid-loop operation with no or small openings,
  - Loss of RHR at mid-loop operation with large openings,
  - Loss of RHR at mid-loop operation with a dam in a hot leg,
  - Boron dilution at shutdown,
  - Cold overpressure transients at shutdown.
- Test types related to Accident Management:
  - High pressure primary side feed and bleed,
  - Low pressure primary side feed and bleed,
  - Secondary side feed and bleed,
  - RCP restart in a highly voided primary system,
  - Primary to secondary leak with multiple failures.

The objectives of validation on IETs are:

- To assess the consistency of the physical model package
- To test overall code performance

– To check the code capability to represent system effects with interactions between components
– To draw attention to points which need further physical investigations
– To give guidelines to users (modelling, nodalization)

Example of system effects which shall be considered specifically in validation are reflux condensation, steam binding and loop seal clearing.

The selection of integral effects tests should consider:

– Design of the facility according to a scaling strategy based on identification of key phenomena and scaling of these phenomena.
– Availability of well-defined test facility design and test conditions (IC &BC).
– Quality, quantity, and availability of the measurements with estimation of uncertainty.
– Well documented test reports and test results.
– Geometry representative of actual reactor.
– Transient scenario representative of actual reactor scenario.
– Scale as close as possible to reactor or scale complementary to other similar data at a different scale which will allow scaling capability of the physical models to be checked.

Some available IET test facilities for western type PWRs (with U-tube SG) are defined in Tab. A4-3 (see chapter 3 for more details).

The analysis of an IET calculation should contain:

– A justification of all choices made when creating the input deck including nodalization and transient management. An identification of key phenomena of the transient and of the measured parameters which are relevant to them.
– A determination of the quantitative or qualitative accuracy of the prediction of key phenomena.
– An analysis of the reasons of the main code-experiments discrepancies using sensitivity tests on the nodalization, on the physical models or even on numerical options:
  - Problems of nodalization
  - Models used out of their domain of validity:
  - Physical process not modelled: (CCFL).
  - Transients highly sensitive: loop seal clearing.

Tab. A-4-3 – Key features of ITF suitable for code assessment.

Test Facility	Height Scale	Volume Scale	Power	Pressure (MPa)	No of Loop	Core type
LOFT	½	1/48	100%	16	2	Nuclear
SEMISCALE	½	1/3000	100%	16	2	Elect.
LSTF	1/1	1/48	14%	16	2	Elect
BETHSY	1/1	1/100	10%	16	3	Elect.
PKL	1/1	1/145	10%	4	4	Elect.
LOBI	1/1	1/700	100%	16	2	Elect.
SPES	1/1	1/427	100%	16	3	Elect.
ATLAS	½	1/288	10%	16	2	Elect.

The NEA ITF data base, see e.g. NEA/CSNI, 1996, included several tens experiments from about 20 test facilities or reactors. Since then numerous new tests were performed which should be considered for validation.

A-4.3.5 Validation on large scale experiments

Since most SETs and all IETs are reduced scale experiments, it was found necessary to add a few large scale (or even full scale) experimental data to check the scaling capability of the models and of the system codes in relevant high ranked physical situations such as Downcomer Refill, Core Reflooding, Upper Plenum CCFL, and Hot Leg injection.

They are not exactly SET since they have a lower density of instrumentation and they are not fully IETs since they do not represent all components of the system.

The international 2-D / 3-D programme included the UPTF tests, the SCTF and the CCTF tests already presented in Chapter 3 (significant references are listed in Chapter 3). They played a very important role in scaling strategy by providing the way to validate key phenomena at the reactor scale.

A-4.3.6 Validation on containment experiments

System codes are also used for containment thermal-hydraulics with a lumped parameter modelling. The whole containment is modeled with one control volume or with a multi-compartment modelling. Specific experiments are used for the validation such as PANDA (e.g. see NEA/CSNI ISP 42), MISTRA, TOSQAN (e.g. see NEA/CSNI ISP 47), HDR (e.g. see NEA/CSNI ISP 23 and ISP 29), THAI (see e.g. OECD/CSNI/NEA ISP 47) and CVTR (Schmitt et al., 1970).

A-4.3.7 Validation on NPP data

NPP data are scarce but they are the only data which actually allow a validation on real geometry, and actual physical condition. In this respect they play a role in the scaling assessment and in the validation of reactor nodalization.

Most NPP data refer to operating conditions or in some operational transients and start-up test. They may include some tests done on the reactors to test the response of particular components (pressurizer or steam generators for example), and a few incidental transients, such as:

- Peach Bottom turbine trip
- pump shutdown
- Kozloduy-6 pump restart

They may include data from a few accidental transients:

- DOËL and MIHAMA SGTR
- TMI 2 accident (early stages of accident before core degradation)
- Chernobyl 4 accident (early stages of transient up to power excursion)
- RBMK single pressure tube rupture incident due to inadvertent closure of a valve (e.g. Leningrad 1992)
- La-Salle 1988 BWR instability event.
- Zaporoshe 1995 Pressurizer stuck open valve accident

Some Steam Generators had additional instrumentation to provide some validation data (recirculation ratio, temperatures …).

A-5 EXTENDED EXECUTIVE SUMMARY

Scaling in nuclear-thermal-hydraulics constitutes the topic of this document. To produce the document, a Specialist Scaling Group (SSG) was formed in 2013 by the WGAMA of the NEA/CSNI; the writing of the document was completed in 2016. The need for this document testifies the importance of scaling in nuclear technology, but also to the controversial evaluations of scaling-related findings by the scientific community.

Clarification of the words or expressions “scaling”, “scaling issue” and “addressing the scaling issue” , and then reaching a consensus on those terms is the first priority among community members. Namely, “scaling” is the process of converting any parameters of the plant at reactor conditions to those either in experiments or in the results of numerical code so to reproduce the dominant prototype phenomena in the model; “scaling issue” indicates the difficulty and complexity of the process, and the variety of connected aspects; and “addressing the scaling issue” is a process of demonstrating the applicability of those actions performed in scaling.

The appraisal of scaling through history (e.g. see Appendix 2) revealed its existence and importance since the beginning of the nuclear era. The impressive amount of scaling research and activities is compiled in Fig. 1-1, the Scaling Database and Knowledge Management. This figure presents three categories of scaling activity.

1. Technological bases to undertake scaling, which include the experimental data, results of analyses results, journal papers, and OECD reports.
2. Requirements for scaling, which include those derived in CSAU, CIAU, BEMUSE, and the system codes, V & V.
3. Techniques and approaches used in scaling analyses, which include scaling methods such as power-to-volume, H2TS, FSA, DSS and the applications of system code.

The goal of the three categories of scaling activity is to address the scaling issues.

Chapter 2 was written with the idea of drawing a picture of the overall scaling- universe in today’s technical community. This chapter surveys as systematically as possible, the commonly-accepted topics and the controversial ones associated with scaling. These topics include scaling distortion, scaling of complex phenomena, and the role of scaling in safety applications and reviews. The purpose is to give the reader a broad background on this SOAR subject.

The chapter begins with an overall picture of the scaling to depict its subjects from several perspectives. The SSG reviewed the activities that prompted the development and application of some milestone scaling-techniques. The second theme is the introduction of the relationship between thermal-hydraulic scaling and nuclear-reactor safety. Finally, some significant achievements accomplished in scaling are captured briefly and highlighted herein.

– Flashing, flooding and counter-current flow limitation (CCFL) in the downcomer of the PWR reactor pressure vessel (RPV) during a large-break loss-of-coolant accident (LBLOCA);
– Wall evaporation, flooding, and CCFL in the downcomer of the steam generator’s (SG’s) secondary side, during accident-recovery conditions;

- Influence of reversed-flow U-Tubes on the natural circulation performance of the SG primary-side flow;
- Simulation of nuclear-fuel rods in integral test facilities (ITFs) using electrically heated rods with and without a gap;
- Concept of scaling distortion in the uncertainty method, based on extrapolating uncertainty;
- Concept of a Scaling Pyramid that summarizes current scaling approaches.

Scaling Distortion

Following other researchers, (a list of citations is provided in Reference Section) the distortion due to scaling is reviewed as an important topic of interest. It is well known that scaling distortion is at the center of all scaling issues. The origin of scaling distortion was explained using some well-known examples in three scenarios, in general: the assumptions and simplifications in scaling methods, the limitations in constructing and operating test facilities, and the scalability issues embedded in the computer codes.

To explore distortion in scaling, most scaling methods are evaluated to illustrate their deficiencies. For example, the Buckingham Pi theorem brings in non-dimensional groups that may or may not have physical meaning directly related to the phenomenon. The other global approaches are based on conservation laws. The local approach could generate many similarity factors, such that all of them cannot be matched simultaneously in the design of the test facility. The illustration of deficiency focuses on the approaches based on conservation laws (global and local), and discusses some well-known deficiencies identified in these scaling methods. The details of these scaling methods are described in Chapter 3.

The advent of thermal-hydraulic computer codes greatly improved the thermal-hydraulic analyses, a feat that could not be achieved with pure analytical methods. In Chapter 4, the merits and deficiencies of scaling aspects of thermal-hydraulic codes are reviewed in detail. However, computer codes raise an important issue: the capability of the code for simulating reactor conditions is usually limited by the scaling distortion that includes a scaling limit. The nature of this issue is that the fundamental physics models in the code are built by correlations and empirical criteria (coefficients) and are included in the code-balance equations. Constants of these empirical formulas sometimes are determined by curve fitting, and may depend strongly on the geometry (shape and size), and fluid conditions.

Furthermore, once the code's applicability has been determined, a statement is needed of uncertainty in the predicted safety parameters. This is estimated through a combination of bias and the distribution of uncertainties for each correlation, and the boundary- and initial-conditions. It is a complex subject to account for the scaling distortion in the uncertainty-evaluation process. The details of the relationship of scaling and uncertainty are reviewed in Chapter 4.

It is well recognized that distortion is inevitable in scaling complex systems, like light-water reactors (LWRs). Scaling laws usually are derived from the dominant physics in each phase of the transient and/or scaling methods. The dominant phenomena will change from one phase to another of the transient, and, with this, the scaling groups also will also change. It is unlikely to reach a perfect similitude between the reference system and the experimental model for all phenomena in one transient. The scaling distortion could become extremely large. In the AP600 scaling analysis, for instance, similarity parameters related to the pressurizer-level transient of plant and APEX facility could differ by 20 times due to size scaling of the surge line. As such, it is difficult to determine the acceptability criteria for distortion in an experiment. The propagation of the effects caused by distortions represents another need to call for a method that can evaluate the accumulated distortion of a process as a function of time.

Another category of scaling issues is the scaling of complex thermal-hydraulic phenomena. Some of these phenomena, shown below, are discussed in Chapter 2. Among these, TPCF and CCFL are reviewed further in Chapters 3 and 4:

- Two-phase critical flow (TPCF),
- Counter-current flow limitation (CCFL),

- Entrainment and de-entrainment,
- Reflood,
- Fuel-rod ballooning,
- Special plant components; pumps, separators, and similar ones,
- Core local phenomena at sub-channel level.

The purpose of reviewing these topics is to identify future challenges in scaling. Most of these phenomena are too randomized and localized to describe with standard governing equations. They usually affect the emergency operation of the core cooling (ECC) system and its consequences, and cannot be neglected in the scaling. Due to their chaotic nature, empirical correlations normally were used to derive the scaling laws. This may pose a great challenge to the scaling capability of the scaling laws so far obtained since the correlations were mostly developed in scaled separate-effect test facilities (SETFs).

Since the start of experimental nuclear-thermal-hydraulics, there is a long history of scaling activities connected to the design of experiments and the construction of the test facilities. The decision of choosing an integral effect test (IET) or a separate effect test (SET) or using both is another challenging topic in scaling. The basics of design and choice between IET and SET and their scaling bases are briefly reviewed. Recently, the use of counterpart tests and similar tests to explore complicated phenomena has become prevalent. Chapter 3 has more examples on this topic.

A Scaling Analysis for the Safety-Review Process

Scaling is relevant to the safety of nuclear power plants, and specific scaling analysis usually is needed in the safety-review process. In Chapter 2, the scaling role in the safety review process is described through illustrating the relationship of scaling with examples of evaluation model development, EMDAP, USNRC, 2005, and with the quantification of uncertainty, CSAU, USNRC, 1989. Uncertainty approaches are extensively reviewed in Chapter 4. However, it is emphasized that the purpose of this description is not to recommend any specific approaches to meet the safety requirements set by any regulatory agencies, but to illustrate the role of scaling in them.

A safety determination of reactor design and operation is done by evaluating the prototype thermal-hydraulic response through data from experiments, and/or computer code calculations. Since it is difficult to use the reference reactor to obtain data, especially for the postulated accidents, simulations of accidents in experiments with scaled test-facilities are inevitable. The scaling technique used to design the test facility is a key element to understanding the validity of experimental data. The core-scaling technology is reviewed in Chapter 3, which covers the scaling methods, and the design of the test facilities and the experiments.

Scaling methods can be categorized by the target phenomena at both the local and system levels. In general, the scaling parameters for a local phenomenon can be derived by applying a dimensional analysis (empirical approach), or dimensionless governing equations (a mechanistic approach). Dimensional analysis, such as Buckingham's Pi theorem, uses conventional non-dimensional parameters. Another empirical approach is to use correlations and models to derive similarity parameters, or to estimate distortions due to scaling. A well-known example is the criterion for the flow regime transition, based on the Froude number. The approach of the dimensionless governing equation is to simplify the governing equations for both the prototype and model by making assumptions; the similarity relationships can be obtained through the coefficients of the non-dimensional terms in the equations.

To preserve the prototype transient behaviors, it is necessary to develop a method of system wide scaling. The main objective is to preserve kinematic- and dynamic-similarities between the prototype and the scaled-down test facility. Most scaling laws are derived from the non-dimensional governing equations, as mentioned earlier in the local method. For ITFs, another level of scaling needs to be completed by preserving the important local phenomena and by reducing scaling distortions as much as possible. The important phenomena and processes can be identified from the phenomena identification and ranking table (PIRT).

Scaling Methods

The major scaling methods reviewed are summarized below. The review of each scaling method focuses on its major characteristics, merits, limitations, and application areas.

1. **Linear scaling** – The key characteristic of this method is to have the same aspect ratio and the same velocity in the model as in the prototype. This approach can excessively increase the influences of gravitational acceleration.

2. **Power-to-volume scaling** – This scaling method conserves time and heat flux in the prototype. Unlike the linear-scaling method, this method preserves the scale of the gravity term. Therefore, it offers an advantage in reproducing the phenomenon in which the gravitational effect is significant. Furthermore, it is suitable to simulate an accident in which flashing occurs during depressurization. It was successfully used to design most of the integral-effect test facilities, such as LOFT, SEMISCALE, LOBI, ROSA-II, ROSA-III, PKL, LSTF, and BETHSY. Also, this method is suitable for the heat-transfer test with electric fuel bundles. However, when it is applied to a smaller facility with the full height, due to the smaller area ratio, some important phenomena can be distorted, for example, the excessive stored heat in structures, and a higher-surface-to volume ratio leading to higher heat losses from structures, and loss of multidimensional-flow phenomena.

3. **Three-Level scaling** – The first step is an integral- or a global-scaling analysis to conserve a single and/or a two-phase natural circulation flow. The similarity requirement is obtained from a 1-D non-dimensional, governing the equations of natural circulation. The second step is a boundary flow, and inventory scaling. The geometry is determined to scale the flow rate at the junction of a broken part, the safety-injection system, and various filling- or discharge-systems in the ITF to ensure the inventory of the mass and energy similarly is preserved in the ITF as a model of the prototype. In the last step, a local phenomenon scaling is performed to conserve the important thermal-hydraulic phenomena occurring in each system. The result from scaling the local phenomenon takes priority if the similarity requirement differs from that derived in the integral scaling. The three-level scaling method is characterized by relaxing restriction on the length scale. By adopting a proper length-scale, some distortion of the flow regime and multi-dimensional scaling in the scaled ITF can be reduced. On the other hand, the scales for time and velocity are lowered due to the reduced length. Consequently, some local phenomenon could be distorted.

4. **The Hierarchical 2-Tiered Scaling (H2TS)** – The procedure consists of four stages, i.e. system decomposition, scale identification, top-down analysis, and bottom-up analysis. At the first stage, the system conceptually is decomposed into subsystems, modules, constituents, phases, geometric configurations, fields, and processes. The scale identification as the 2nd stage provides the hierarchy for the characteristic volume fraction, spatial scale, and temporal scale. To establish the hierarchy of the temporal scale, the characteristic frequency of a specific process is defined, and then the characteristic time ratio can be found by dividing the system's response time based on a volumetric flow-rate. The top-down scaling as in the 3rd stage offers a scaling hierarchy, using the conservation equations of the mass, momentum, and energy in a control volume. In the non-dimensionalized balance equations, the characteristic time ratio represents a specific transfer-process between constituents. All the processes can be compared, and ranked for importance on the system to establish priority in the scaled models. The bottom-up scaling, as the 4th stage, offers a detailed scaling analysis for key local phenomena, such as the CCFL and choking. Along with this top-down analysis, similarity groups (called Pi groups) are identified, and the scaling criteria and time constants can be obtained to evaluate the relative importance of the processes.

5. **Power to Mass scaling** – To determine the test conditions for a reduced-height and reduced-pressure (RHRP) facility, the power-to-mass scaling method, was developed. This method determines scaled core-power according to the initial coolants' mass inventory in the reactor's coolant system. The temperature of the hot leg in the test facility is determined from the sub-cooling of the primary system, which is made the same between the model and the prototype. From the hot-leg temperature, the cold-leg temperature is determined by the equivalence of the core temperature's difference for the model and prototype. The mass flow rate of the core is scaled down according to the power and heat capacity relationship. Finally, secondary system pressure is determined from the difference in temperature between the primary- and the secondary-side. Since pressure is not preserved, the differences in fluid's thermal properties could induce distortions.

6. **Modified Linear scaling** – The multi-dimensional behaviors of the Emergency Core Cooling (ECC) water in the downcomer (e.g. the ECC bypass) are observed during the LBLOCA refill-phase. The modified linear scaling method was developed to overcome this distortion in a small-scale test facility. Twelve dimensionless parameters were obtained from the two-fluid momentum equations in the downcomer. By preserving those parameters in the model, the method resulted in the same geometric similarity criteria as in the linear scaling method. However, this method conserves the gravity scale. It also was found that the three-level scaling method provides the same requirements when the area aspect ratio is preserved as square of linear ratio in a test facility.

7. **Fractional Scaling Analysis (FSA)** – FSA is a hierarchic approach similar to H2TS. In the first step, the regions of interest and the durations of the transients are specified. The rate of change of the state variables over the region are connected to the transfer functions defined at the boundary, and inside the volume. The relative effect of components is based on their relative impact on state variables in the transfer function connected to that component. These relative values determine the importance of these transfer terms. The fractional change-of-state variable (effect metrics) over the characteristic time (fractional change metric) should be made the same between the prototype and its model in top-level scaling. The characteristic time is obtained either from the experiments, or from an aggregate fractional rate of change (FRC, also called the aggregate frequency). The individual FRC can be positive or negative. The reference value of the agent-of-change should be the maximum value over the period of the phase. FSA offers a systematic method of ranking components and their phenomena in terms of their effect on the figure of merit (FOM), or the safety parameter. It also can estimate scale distortions, and synthesize data from different facilities for the same class of transients. This multistage scaling can guide the design, and simplify the scaled facility by identifying important components and corresponding processes. This approach does not require the preservation of time.

8. **Dynamical System Scaling (DSS) –** To address the time dependency of scaling distortion, an innovative approach was developed recently. The strategy is to convert the transport equations into a process space through a co-ordinate transformation, and exploit the principle of covariance to derive similarity relationships between the prototype and model. After the transformation, the target process can be expressed in the process-time space as a three-dimensional phase curve, called geodesics. If a similarity is established between model and prototype, these two phase-curves will overlap at any moment of the transient. Any deviation of the process curves represents the deviation of scaling as a function of time. By specifying the ratios of the conserved quantity and the process (called 2-parameter transform), the generalized framework can be converted to a specific scaling method, such as the power-to-volume scaling. Furthermore, this generalized approach offers the benefit of identifying the distortion objectively and quantitatively at any moment of the transient.

Depending on the objectives of an experiment, as well as the constraints due to such conditions of budget, facility building size, scaling approach is applied for the scaling of height (volume), time and/or pressure. It is usually applied during the preliminary stage of test facility design. The table A5-1 briefly compares the advantages and disadvantages of phenomena scaling.

One criterion that should be considered in designing the scaled-down facility is to maintain the minimum dimensions to preclude some size effects that would not occur in the prototype, for example, the surface tension effect: Boucher et al., 1990, established the concept of a minimum dimension from flooding considerations, and a dimensionless diameter was established. As long as the dimensionless diameter is greater than approximately 32-40, the geometry is sufficient to preclude the influences of surface tension. Other criteria related to hydraulic resistance (friction numbers), stored heat, and heat loss need also to be considered.

Scaling methods are essential tools in the nuclear thermal-hydraulics. However, the scaling methods alone are not sufficient to address the needs in quantifying the safety margins. At least four drawbacks or limitations can be identified:

1. Choice of starting equations.
2. Approximations in selecting non-dimensional numbers for scaling some local phenomena.
3. Details of geometry and the initial conditions of the NPP.
4. Local validity.

Therefore, the experiments are indispensable to complement the scaling methods to address the safety margins and uncertainties in the safety of nuclear reactors.

Role of Experiments in Scaling

The experiments in nuclear thermal-hydraulics can be grouped into three categories: basic tests, Separate Effects Tests (SETs), and Integral-Effect Tests (IETs). Basic tests aim at understanding the phenomena and do not make necessarily reference to the geometry or to the actual ranges of operating parameters in power plants. Therefore, basic tests have a weak connection with scaling. Examples of this type are friction pressure drops, heat transfer, sonic speed for single- or two-phase flows, mixing of hot and cold fluids, and direct contact condensation.

SET, as the 2nd category, is designed to observe phenomena in selected zones in a nuclear-power-plant's system or in specific plant components and some specific process in a particular period of a given transient. The major role of SET is to provide experimental data to develop and validate the physical models, and/or empirical correlations under prototypical- or simulated-conditions. Recently, heavily instrumented SETFs were built to produce spatially and temporally fine-resolution data (called the CFD-grade experiment) for validating the CFD codes.

Table A5-1 – Sample comparison of different scaling approaches.*

Approaches/ Design factors	Height scaling (**Full height versus Reduced height**)	Time scaling (**Preserved vs. Reduced**)	Pressure scaling (Operation) (**Full vs. Reduced**)	Loop scaling (**Full prototype number vs. Lumped**)
Space/cost	**Full**: Higher **Reduced**: Lower	**Preserved**: Higher **Reduced**: Lower	**Full**: Higher. **Reduced**: General savings in construction and experiments. May represent low-pressure transients	**Full**: Higher. **Lumped**: Considerable savings in constructing the facility.
Time scale	**Full**: Preserved. **Reduced**: Reduced but	**Preserved**: Possibility of attaining the same	**Full:** Preserved. **Reduced:** Distortion	Not related

Approaches/ Design factors	Height scaling (**Full height versus Reduced height**)	Time scaling (**Preserved vs. Reduced**)	Pressure scaling (Operation) (**Full vs. Reduced**)	Loop scaling (**Full prototype number vs. Lumped**)
	still possible to preserve time by increasing flow resistance.	timing of the event and the local thermal-hydraulic response. **Reduced**: Speed up of slow transient.	may appear during the phenomena transition with phase change.	
Heat transfer phenomena	**Full**: Possibility to reproduce local phenomena, such as CHF, PCT, and re-flood, as well as phase separation. **Reduced**: Short length of heating may cause distortions.	**Preserved**: Possibility of reproducing mass & energy distribution and heat-transfer responses. **Reduced**: Distorted due to time-dependent effects.	**Full**: Possibility to reproduce phenomena up to prototype pressures. **Reduced**: Some phenomena are distorted if using different fluid.	**Full**: Possibility to reproduce mass and energy distributions in loops and core. **Lumped:** Possible distortion in mass & energy distribution at core's entrance (3-D).
Gravity/ fluid acceleration	**Full**: Driving force by gravity preserved, good for natural circulation. **Reduced**: Some gravity-driven phenomena distorted	Same as left comparison.	Not related.	**Full**: Better replication of BICs for asymmetric NC, e.g. Counter drive for NC during SGTR in 3- or 4-loop PWRs
Flow regime/ 3-D phenomena	**Full:** With very small volume scaling, limitations in multi-channel scaling, distortions in multi-D phenomena. **Reduced:** Improved	Not related.	Not related.	**Full**: Good to represent BICs for NC, e.g. PTS at MSLB, re-criticality for boron dilution. **Lumped**: Good for 3-D phenomena in horizontal legs
Experiment design & control	**Full**: Real time operation and control. **Reduced**: Easier operation and control for slow transient in prototype.	Same as left comparison	**Full**: Not particular **Reduced**: Needs interpretation via scaling method. More flexible for measurements of instrumentations.	**Full**: Precise realization of BICs to represent multi-loop response **Lumped:** Limited spectrum of possible asymmetries.
Stored heat/ heat losses	**Full**: Distorted in a very small-volume test facility due to high wall-surface to volume ratio (S/V). **Reduced**: S/V could be reduced.	Not related.	**Full**: Higher than prototype **Reduced**: Closer to prototype due to thinner RCS walls.	**Full**: Higher compared to lumped loops with identical scaling factors. **Lumped:** Improved
Coolant property	Not related.	Not related.	**Full**: Prototype property possible **Reduced**: For fluid-to-fluid simulation, nonlinearity may	Not related.

Approaches/ Design factors	Height scaling (**Full height versus Reduced height**)	Time scaling (**Preserved vs. Reduced**)	Pressure scaling (Operation) (**Full vs. Reduced**)	Loop scaling (**Full prototype number vs. Lumped**)
			appear. Difficult to fully match similarity parameters. Some phenomena not possible.	

Note 1: This table is a high- level sample comparison. Refer to the main text for complete details. Scaling approach of an experiment depends heavily on its objectives.

Note 2: BIC: boundary and initial conditions, NC: natural circulation, RCS: reactor coolant system

The last category comprises the Integral Tests (or Integral Effect Tests) with ITF. ITF is a test facility to provide a dynamic and similar thermal-hydraulic response that may appear in postulated accidents, and/or abnormal transients in the reference reactor. The data obtained from scaled ITF experiments are considered not directly applicable to full-scale conditions. Instead, the data is used mostly in validating the system codes and understanding of accident phenomena. Scaling activities are essential to the design and the operation of ITF.

A comprehensive appendix is prepared to summarize the key parameters of the major test facilities in the world, mainly the ITF and selected SETF. The facility types considered are ITFs for each of PWR, BWR, VVER, advanced reactor and containment, with selected SETFs. Major design parameters and scaling information are given for reference. A synthesis of the information is organized in Table A3-0 in Appendix 3 for quick reference. The focus is on how scaling has been considered in the design and the experiment results.

The main characteristics of SETF for reactor systems are as follows:

1. Minimum scaling distortions by employing full-scale and/or prototype fluid conditions.
2. Dedicated instrumentation to characterize selected phenomena.
3. Well-imposed boundary conditions (B.C.) necessary to simulate interactions with other reactor components

The scaling distortions are mainly due to scaling of external boundary conditions causing a distortion on the interacting phenomena at the facility boundary. In many cases, the data obtained from a SETF can be applied to the full-scale prototype, although the direct extrapolation at the full-scale prototype requires caution, considering the facility’s scaling limits. It is foreseeable that the influence of the facility’s scale is observed in many SETF tests, and the LOCA phenomena easily are influenced by 2-D/3-D effects. Therefore, a full-scale SETF such as UPTF is valuable to characterize multi-D phenomena. In addition, counterpart tests for the same phenomenon l also will provide confidence in extrapolating uncertainties to a full-scale plant.

An ITF for reactor systems can be characterized as a test facility composed, at least, by a heat source and a heat sink connected in a closed loop by hydraulic paths. Several systems can be connected at this closed loop. Considering the scaling approaches used to design an ITF, several distortions might cause the partial- or total-failure of the intended phenomena simulation – called scaling limits. In general, the data obtained from an ITF cannot be directly applied to a full-scale prototype. Such direct extrapolation requires caution in considering the facility’s scaling limits and appropriate methodology. The ITFs of PWRs and BWRs are compared and summarized in this report. An important feature of the VVER reactor is the horizontal steam generator. Therefore, the main heat-exchanger’s characteristics at the facilities are reviewed. Some passive- and advanced-designs for water-cooled reactors are characterized by new phenomena, and accident scenarios such as containment phenomena and interactions between the containment and reactor coolant system (RCS), low-pressure phenomena, and the phenomena of new components or reactor configurations. Traditional scaling approaches and methods are applicable for considering these advanced features in the design.

For the SETF of the primary containment vessel (PCV), the scope of the present SOAR does not include a scaling review of severe accidents. In addition, each SETF corresponds to specific situations, depending on the design's objectives. Therefore, the review focused on some highlights on the scaling techniques related to the DBA phenomena. Different containment designs for each reactor type are compared, which include PWRs, BWRs, VVERs, and new NPP designs, such as small modular reactors (SMRs).

For the PWR PCV-ITF, due to the fact that earlier existing facilities were a part of small yet real power plant able to provide experiments and data, the scaling analyses received limited attention. The interest on scaling for the PCV-ITF arose when discrepancies were observed in the results between HDR (Heissdampfreaktor) and BFC (Battelle-Frankfurt Containment). In the PCV-ITFs, the 'expected time of event' is preserved when the addition of energy and mass from the reactor's primary system and the subsequent distribution along the several compartments of the containment is preserved. The facility's material and the relative spatial distribution and proportion are important in designing facilities. Other scaling characteristics of the PCV-ITF are the compartmental subdivisions and energy-release scaling into PCV.

The scaling compromise is one of the major reasons to cause scaling distortions due to the difficulty of complete similitude in all local phenomena and even the lack of knowledge of the local phenomena themselves. In this review, the following main scaling distortions observed in the experiment are identified as follows:

1. Circular sections – due to hydraulic diameters not being preserved.
2. Structural heat loss and stored heat – due to a larger structural mass and structure surface area per unit coolant volume relative to the prototype.
3. Inventories and inter-component flows – due to choked flow.
4. Pressure drop – due to the ratio of length divided by diameter is very large.
5. Multi-dimensional phenomena – due to tall and skinny nature to preserve power volume and height.
6. Scaled-down reactor coolant pump – reliable two-phase pump model is not available until now, specific speeds and single-phase characteristics are recommended to be preserved.
7. Fuel simulators – electrically heated fuel simulators may behave differently from nuclear fuel rods.
8. Scaling distortions of local phenomena – due to inherent scaling distortions by design and simulation constraints, and non-typicality of local phenomena.

It should be noted that not all local phenomena are of equal importance in influencing the FOM or the parameter of interest. The global scaling approach provides that guidance.

Counterpart Test (CT) and Similar Test (ST)

As data acquired in experiments at a single (scaled) test facility may be questionable due to inherent scaling distortions, the concept of counterpart tests (CTs) involving several ITFs or SETFs at different scales and design concepts have been considered important. It is desirable that the following minimum set of BC/IC values and related parameters are preserved between the CTs.

1. Thermal-hydraulic state and parameters (pressure, temperature, and flow condition) in each component of the facility:
    a. Scaled values to power-to-volume scaling ratio (kV).
    b. Characteristics of primary- and secondary-side safety and operational systems (e.g. accumulator-injection and safety-injection systems, SIS, characteristics).
    c. Heat- and mass-sinks or sources (e.g. location and size of break).

d. Timing of operator's actions based on pre-defined operational criteria.

Good examples of CT include the SBLOCA tests by LOBI, SPES, PSB, BETHSY, and LSTF. All five test facilities simulate the primary circuit of a PWR (VVER in the case of PSB and Western PWR for the other facilities) with original heights covering a broad range of volume-scaling factors: LOBI: 1:712; SPES: 1:427; PSB: 1:300, BETHSY: 1:100; LSTF: 1:48. The similarity of the overall results confirms the choice of the adopted scaling laws and the suitability of the individual test facilities to reproduce a plant's typical behavior under the given BCs. The CT tests conducted between PKL and LSTF (between the NEA PKL-2 and ROSA-2 projects) demonstrated the effectiveness of a secondary-side depressurization in removing the heat from the primary side, which achieved almost identical primary-depressurization behaviors, which enabled a systematic comparison of the thermal-hydraulic response between the two ITFs.

ITF experiments whose BC/IC were not aligned according to the requirements of CT are referred to as "similar tests" (ST). These experiments demonstrate that the differences of CT and ST lie mainly in the BC/ICs. A special group of tests called complementary tests where, in the same set up, the ITF concentrates on studying the overall system's response and the SETs investigate the responses of the plant's subsystems and phenomena which are highly dependent on the geometry (in scales up to 1:1 full-scale, such as UPTF). Another category of tests referred to as daughter (facility) tests employs results available in 1:1 full-scale as the reference for a comparison with the results from scaled-down experiments on the same phenomena. It aims at evaluating the scalability of relevant phenomena and their understanding in general.

Role and Characteristics of the System Code

System codes incorporate the knowledge obtained from the available large data base. The system codes also can help in analysing complex transients. In addition, system codes can perform calculations related to scaling. The merits and limits of codes related to scaling are reviewed during code development, code verification, and code validation. A few sections (more details are provided in Appendix A-4) are devoted to an overview of the governing equations to give readers a background in considering the effects of scaling.

In the process of developing code, several averaging simplifications are made on the space- and time-scale of the processes. Some distortions are introduced due to simplifications of the physics, non-modeled phenomena, and the limited accuracy of the closure laws. Therefore, several inherent limits are summarized as

– **Space and time averaging**: System codes do not predict small-scale thermal-hydraulic phenomena due to space averaging and cannot predict all the small time-scales associated with turbulence and two-phase intermittency.
– **The dimensions of the model**: Using the O-D (or lumped) model, 1-D models, or a porous 3-D approach consists of simplifying a complex 3-D flow; using a 1-D heat conduction in heating structures and in passive solid structures is an approximation for more complex 3-D conduction.
– **Flow regime maps**: The highly empirical flow-regime maps are valid only in some states, such as the steady state, the quasi-steady state, the fully developed and the quasi-developed states; while the rapid transient- and non-established-flows could exist in accident conditions. The flow regime also depends on geometry, conduit size, and the fluid's physical properties. These differences from established flow regime maps should be introduced in the numerical model.
– **Scaling of each closure law**: Closure laws in system codes may be purely empirical, mechanistic, or semi-empirical. Therefore, the scalability of the closure law is questionable.
– **Non-modeled phenomena**: System codes neglect many complex phenomena, as described earlier.

Hence, the up-scaling capabilities of a system code depend mainly on how well it predicts phenomena in scaled SETs and IETs. The scalability needs to be confirmed during the process of validation.

Although code verification is mainly related to the numerics of a code, and the relation with scaling is not strong, the verification does include some aspects related to scales. The codes should be able to resolve the time- and space-scales of the phenomena. The same accuracy and but a unique solution of two similar systems should be reached. Coding errors affect scaling as it may induce some type of error.

The code validation plays a very important role to assure the scalability of code. The following objectives, to assure the scalability, are important in the validating the code with various scaled SETs and/or IETs; viz., closure laws, choice of appropriate code module to model each reactor component with options, assumptions, and modes of calculation, reactor-accident transients, and scaling distortions.

Scaling should be considered in developing the nodalization. Some considerations are needed due to the scales in the system. For instance, it is impossible or impractical to preserve the L/D when setting up nodalization for differently scaled facilities. Choosing a reasonable size of the averaging region, or control volume, is important for acceptable numerical solutions. The arrangement of node density or the size of a control volume could create non-fully developed flow regions (or even stagnated ones) which may not compatible with the facility. Finally, a specific scaling qualification is needed for the K-factors at geometric discontinuities in a nodalization. It is essential for both the levels of steady-state and transient so to address scaling-related concerns, such as what procedures and criteria an analyst should follow to pass the nodalization from the scaled facility to a NPP, and under what conditions can the uncertainty derived in scaled facility remain acceptable under NPP conditions. An approach called Kv-scaling briefly is introduced here as an example for accomplishing proper nodalization.

A Kv-scaled calculation is a procedure for system code simulation procedure in which well-defined (measured) scaled ITF- are converted to an NPP-nodalization, and the test is simulated with this nodalization. The purpose is to reproduce, by sensitivity studies, same phenomena as seen in ITF by the NPP nodalization; namely, number of nodes and node sizes are changed in the framework of the process. Performance of both NPP and ITF nodalization can be compared to check the validity of the NPP nodalization for any needed corrections and improvements. The procedure is systemized to qualify NPP nodalization.

The system code can be used in the preliminary verification of the scaling laws, although it is inevitable to have scaling distortions due to compromises in design or construction. To study these distortions, Ransom et al., 1998, devised a triad method, somewhat reflecting the Kv-scaled method, to relate the scaled experiment to the prototype system. The method is based on three separate, but related system-code models: (1) The prototype; (2) an ideally scaled model; and, (3) the actual scaled experiment. These three models are created to investigate the degree to which qualitative- and quantitative-similarities are maintained among the three systems in a particular process. The benefits of this triad of models are to ensure homology and to ensure (1) the response of the prototype and the ideally scaled model are comparable to assure their qualitative- and quantitative-similarity; (2) the effect of any experimental non-typicality, such as physical configuration, heat loss, real valves' opening times are evaluated via the scaled model and the prototype.

Scaling in Uncertainty Methods

The relationship of scaling and the uncertainty method is another important subject in this review. The purpose is to show how scaling is quantified as a source of uncertainty in the prediction of NPP transient. Here, we review three uncertainty methods – CSAU, UMAE-CIAU, and the GRS Method.

In the CSAU procedure, three uncertainty sources are quantified as follows: (a) The code and experiment accuracy, (b) the effect of scaling, and, (c) the reactor's input parameters and state. The first two are normally combined. A scaling study is performed based on such information as the PIRT results and the code assessment manual. With this information, uncertainties and biases are determined based on the following two sources as:

1. Evaluation of scaling distortion from test facilities of different scales in the same important phenomenon;
2. Evaluation of scale-up capabilities of closure correlations used in the code.

All available scaled data used to develop the correlation or model in the code are compiled to determine the uncertainty or bias so to reach the 95% confidence level. Additional biases are needed if the range of NPP conditions is not covered in the tests. After evaluation, all the uncertainties and biases are added together as the total uncertainty in the FOM.

In UMAE, experimental data is related to the corresponding calculated results, and an 'error-scaling' procedure is performed. Therein a database is constituted by time trends of the relevant thermal-hydraulic parameters measured in ITFs with different scales and their 'qualified' code calculations. As some conditions are met, e.g. a sufficient number of experiments in different scales and the error of prediction is not scale-dependent, then the error which shows a random character can be extrapolated to the NPP' conditions. A key scaling step of UMAE is the similarity between the NPP prediction and one set of ITF experimental data. This state is achieved through the Kv-scaled calculation.

The GRS method is a widely used uncertainty method based on probability calculus and statistics. The main advantage to using these tools is that the number of calculations is independent of the number of uncertain parameters to be considered. The necessary number of code calculations is given by the Wilks' formula, which depends only on the chosen tolerance limits, or the intervals of the uncertainty statements of the results. The method requires first identifying the important phenomena (PIRT), and then the potentially important contributors to the uncertainty of the code results. Uncertainty due to scale effects is one of them. The probability distributions of each uncertainty must be quantified. After qualification process is done for code, and the nodalization is established, the combination and propagation of uncertainties is executed. Finally, the scale-up effects in the method are evaluated by quantifying model uncertainties in facilities of different scales and uncertainties due to input.

Scaling Roadmaps

Scaling roadmaps are discussed which focus on the design of experimental facilities (Chapter 3, section 3.4), and on the nuclear reactor's safety assessment (Chapter 4, section 4.5). One of the scaling roadmap for designing test facilities is based on the DSS method already discussed. Details are given below for scaling roadmaps for safety assessments.

There is a need to address scaling issues in a safety-review process using the available data, tools, methods, and approaches. A scaling roadmap is proposed to group these actions and information. Due to the different approaches of BEPU methods, there are different ways to meet the safety requirements. Two scaling roadmaps are provided for the reader's reference.

A generic scaling roadmap is proposed, first based on CSAU. A diagram is shown in Fig. 4-21. In this map, a scaling method is chosen to design test facilities at smaller scale to simulate the phenomenon of interest which is expected to occur in the NPP. These test facilities provide essential information for designing the plant, and for assessing the efficacy of safety systems. With the data, the expected thermal-hydraulic processes and phenomena of power plant can be simulated through calculations with the system code. The results obtained are evaluated by regulators. The fidelity of predictions is estimated by aggregating the contributions of uncertainties from the code models, nodalization, numerics, user options, and approximations of the power plant's representation.

Another scaling roadmap was proposed by D'Auria & Galassi, 2010, described in Fig. 4-22. In this approach, most elements in the Scaling Database and Knowledge Management (Fig. 1-1) constitute the major steps. Differences from the previous roadmap are that some qualitative- and quantitative-acceptability thresholds are embedded in the major steps. These safety requirements either are established by the regulator or first proposed by the licensee and accepted later by the regulator. Non-compliance of safety requirements leads to halting of the procedure and requesting for additional calculations, experiments, and/or R & D.

Role of CFD Tools for Multi-dimensional and Multi-scale Phenomena

When multi-dimensional effects play a dominant role in a safety issue or a design issue, system codes cannot be used with sufficient confidence, and 3-D CFD tools become valuable. Most of the single-phase CFDs are related to turbulent mixing problems, including temperature mixing, mixing of chemical components in a multi-component mixture (boron in water, hydrogen in gas) and temperature (density) stratification. Two-phase CFD is much less mature than single-phase CFD, but significant progress has been made in the past decade.

Different scales are involved in complex reactor thermal-hydraulics, and it is natural to investigate them with simulation tools at different scales. Three different 3-D simulation approaches can be classified as follows:

1. CFD in porous medium: This scale is dedicated to design, safety and operation studies for reactor cores and tubular heat-exchangers. The minimum spatial resolution is fixed by the sub-channel's size (scale in centimeters) in the sub-channel analysis codes.
2. CFD in open medium: The average scale is millimeters or less. It includes turbulence modelling, using the RANS approach and new approaches similar to the LES in some flow regimes. It also is the only scale that, in principle, can predict the fluid temperature-field, thermal shocks, or thermal fatigue.
3. Direct Numerical Simulation (DNS): In the absence of any space- or time-averaging, the characteristic length may be less than a micrometer. The use of DNS will help in understanding local flow phenomena and developing closure relations.

Both porous medium and open medium RANS can be used for design, safety and operation studies. DNS, due to high level of discretization could be dedicated as a numerical experiment related to separate effects.

Conclusions

The purpose of a state-of-the-art review is to survey the status of scaling technology from different perspectives. However, the technology continues to evolve, and new methods and approaches are being developed. Therefore, it is not appropriate to draw specific conclusions. A few broad conclusions are summarized as follows:

1. The information in scaling studies, namely the experimental database, is available for most reactor types but has not been fully exploited.
2. Scaling methods and models are available for specific targets or objectives. The application to a generic objective may suffer from the limitations of these methods
3. Many non-dimensional scaling groups are derived in scaling methods and models: Knowing the hierarchy of these groups is important when applying scaling methods.
4. Distortions cannot be avoided in any reduced-scale experiment where transient two phase flow is involved. Even in the case of single-phase conditions phenomena, like stratification and entrance effect, may induce distortions in scaling, particularly in passive systems.
5. The impact of scaling distortions upon the performance predicted for any reference system, prototype, or reactor, remains difficult to quantify.
6. Data from scaled experiments cannot be directly extrapolated to the reactor in most cases dealing with two-phase flow.

7. Use of a suitable existing scaling method or development of a new method for a specific experiment is essential in minimizing scaling distortions.
8. The use of a well validated and verified SYS TH code can support any scaling analysis, including checking the scaling hierarchy, evaluating the impact of scale distortions, and correcting the distortions in reactor applications. For a safety determination of an NPP, the application of SYS TH codes can support, but not replace the formal scaling analysis, and is the best tool for up-scaling to the reactor transient of interest after the two following requirements are met (i.e. items 9 and 10 below).
9. Uncertainty from scaling should be accounted for in the overall uncertainty when the SYS TH code is used in predicting the thermal-hydraulic phenomena in NPP accident scenarios.
10. Accurate evaluations of scaling uncertainty in the validation results, model correlations, numerical schemes, and nodalizations are needed to meet the requirements of nuclear reactor safety.

Recommendations

Based on the key findings in each chapter, the recommendations are summarized here without prioritization, for planning future activities.

1. To resolve a safety issue related to a postulated reactor accident, the most reliable approach should combine the use of PIRT analysis, scaling analysis, analysis of a wide SET and IET experimental database (including counterpart tests), and the use of a system code in a BEPU approach. In some cases a multiscale simulation using CFD tools may provide better insights into local 3-D phenomena.

2. The capability of SYS TH codes to predict facilities of different scales is needed to evaluate the safety of light water reactors (LWR). The recommendation is to include the scalability requirements in SYS TH code validation. The counterpart tests will also be important asset for validating scalability of the codes.

3. The database of existing SETF and ITF CCVM (see Chapter 4 for references), should be extended to include possibly data related to advanced reactors (including those using passive safety systems), radial transfers due to diffusion, dispersion of momentum and energy, and cross flows in the core.

4. There is a need for well instrumented tests for validating CFD codes for the water cooled reactors in relation to mixing problems, such as boron dilution, MSLB, PTS, thermal fatigue, or mixing with buoyancy effects in some passive systems, to be considered in the general TH validation matrices. CFD codes must first be validated on single phase tests at different scales.

5. There is a need to identify a qualitative and quantitative framework (precision targets) to judge the quality of a scaling approach. This step is connected with the acceptance criterion for scaling distortions, and with the quantification of uncertainty due to scaling.

6. Full height scaling with suitable flow areas (and volume) are recommended for experimental simulation of passive system, wherein the important phenomena are the boiling and condensation processes, and buoyancy effect due to density change. Full height will provide an accurate characterization of phenomena such as natural circulation and related stability.

7. Specific scaling related training is worthwhile in a number of contexts. On both the industry and regulatory sides, good training and education of safety analysts should include, in addition to basic single phase and two-phase thermal-hydraulics, advanced topics of scaling techniques, identification of the dominant phenomena of major transients, code V & V and UQ requirements and code scalability requirements.

8. Revisiting systematically the scalability of system codes at the basic level of each closure law may be a good exercise for training new code users, so to improve the understanding of code scaling, uncertainty and to improve code documentation.

Multiscale analysis using several numerical tools at different scales will help in future to provide more accurate and reliable solutions to reactor issues. This approach requires first that the capabilities and limitations of 3-D two-phase flow calculation (CFD) methods for flows relevant to an NPP are well identified. The simulation capability of details of local phenomena aiming for a replica of the phenomena must be improved. Up-scaling methods for modelling should be developed to use small-scale simulations for improving the closure laws used in SYS TH codes. The CFD tools also should follow an appropriate process of code validation to prove their capability for extrapolation to the NPP-prototype phenomena.