Modeling, Validation, Scaling, Uncertainty and Application in Nuclear Thermal Hydraulics

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ABSTRACT

Nuclear Thermal Hydraulics (NTH) implies a universe of knowledge, which, among the other things, is not easy to summarize and can be described (correctly) from different viewpoints. Resulting pictures are not necessarily consistent among each other and may prevent an optimal planning of researches, definitely slowing down the progress in understanding. Current generation and forthcoming water-cooled nuclear reactors are at the center of the attention; however, a few statements in Conclusions deal with thermal hydraulics in 'other' types of reactors. Rather than suggesting way-outs in future investigations, an interpretation of the state of the art constitutes the target for the paper. Referring to the words in the title, we try to substantiate the following evaluations, obvious or well established: (a) modeling is imperfect; (b) validation is essential; (c) scaling is controversial; (d) uncertainty is a guru-type of matter; (e) applications need proper strategy and consideration of research outcomes, including connections with different technologies. The Best Estimate Plus Uncertainty (BEPU) approach embeds the topics above and fixes the links between NTH on the one side and the Probabilistic Safety Assessment (PSA) and licensing on the other side.

KEYWORDS
Nuclear thermal hydraulics; Modeling; Validation, Scaling; Uncertainty.

1. INTRODUCTION

As a provocative statement, one may state that lack (or uncertainty in) of knowledge in Nuclear Thermal Hydraulics (NTH) brings to the perception of weak capabilities for computer codes, then unreliable results for deterministic assessment and, finally, questionable (and unacceptable) reactor safety. Thus, the acceptability of current large reactors depends on NTH. Obviously, this is not true: design of reactors, quality of construction and organization of operation mainly determine the safety and NTH has an ancillary role. However, the mistrust towards capabilities and uses of the thermal hydraulics computer codes flows around scientists and decision makers and may contribute to the (provocative) statement.

Two-phase thermal hydraulics constitutes a unique subject in nuclear technology: (a) rigor in modeling, e.g. characterizing elasticity theory or neutron physics, is missing; (b) cost of experiments can be huge and the measured data unavoidably come into the scaling issue; (c) it has a pivot role, e.g. connected with other subjects, in demonstrating the safety of nuclear reactors. Developments during the last two decades resemble the movement of the boundaries of a swamp where directions, or way-outs for progress, are weak or even lost.

Water Cooled Nuclear Reactors (WCNR) are within the scope here, restricted to the conditions of core keeping its geometric integrity, i.e. following hypothetical Design Basis Accidents (DBA). Thermal hydraulics in Small and Modular Reactors (SMR) using water as coolant is also of concern. However, accident scenarios with degraded core (i.e. Severe Accident, SA) and non-WCNR are out of the consideration even though NTH encompasses those conditions and systems: as it may result clear from the paper, research investments and maturity of techniques in applications are largely different, i.e. in case of SA or non-WCNR related to WCNR-DBA.

Although this is ambitious or even useless, the objective of the paper is to provide an interpretation for the current state of art in NTH, distinguishing the sectors of Modeling, Validation, Scaling, Uncertainty and Application. The investigations completed or in progress, [1], [2] and [3], constitute the background. The framework role of Probabilistic Safety Assessment (PSA), [4], insights into the past three-decade's history, [5], and streamlining possible way-outs for research, support the discussion and the structure of the paper. Namely, the evaluation of the foundation processes for NTH at the early age of nuclear technology, i.e. the 50's or the early 60's during the last century, has allowed the proposal for research directions with the awareness that feasibility is limited by the need to produce results in short times, [6].

NTH is a self-standing discipline; however, the word 'nuclear' imposes referring to reactors and foreseeable transients. Then, the role of PSA is evident: system design and situations expected in case of accidents are relevant. Multi-links between NTH and PSA occur and low-probability events fix the boundaries for the scope of investigations, [4]. With different words, the predictability of the turbulence generated by the pylons of a bridge immersed into the river is of no interest for NTH (is this right?).

The achievement of conceptual safety demonstration of existing reactors towards the end of last century brought to slow-down in investments, namely for experiments. Corresponding slow-down in improvements in predictive capabilities for NTH system codes is a result, [5]. Furthermore, the (urgent) requests made by USAEC in 1971, [7], and the needs derived from Three Mile Island Accident in 1979, [8], catalyzed the resources for research and prevented, or limited, deep digging into the bases of modeling in thermal hydraulics.

The reader may not expect new investigations or ideas from the present paper; rather, we provide critical insights and interpretation of controversial issues distinguishing the sectors part of the title.

2. MODELING

We refer to the set of Partial Differential Equations (PDE), which are at the basis of system thermal hydraulic codes. However, we distinguish between equations derived from first principles of thermodynamics and mechanics, closure equations and special models.

2.1. Balance Equations

Navier-Stokes (N-S) equations are of concern and are at the basis of the PDE adopted in thermal hydraulic system (SYS TH) codes. The PDE form the Unequal Velocities Unequal Temperature (UVUT), 1-D model, with properly time and space averaged unknowns (e.g., $p, \alpha, w_g, w_f, T_g, T_f$). Figure 1 deals with an outline history focusing on NTH. Books [9] and [10] discuss the features, the merits and the challenges of the N-S equations from the mathematicians and NTH applications viewpoints, including CFD and DNS. Book [1] (and [2]) discusses the equations at the bases of SYS TH codes.

The motion of single-phase incompressible fluid constituted the scope for the original Euler equations considered by Navier (a civil engineer); Stokes (a mathematician) introduced the viscous term and the N-S equations appeared. The concepts of Mass, Momentum and thermal Energy (MME) transfers between 2 phases (or 2 fluids) were far away the initial formulation. In two-phase flow, a force may cause a velocity change that induces pressure change and phase change; in case of water, flow configurations (or flow regimes) appear in experiments and affect MME transfers, i.e. not the situation of mechanics of continuum, [11]. In SYS TH codes, the solution of the PDE is possible in combination with 1D or 3D conduction heat transfer (or conjugate heat transfer modeling) in solids and, preferably 3D, neutron physics in nuclear fuel.

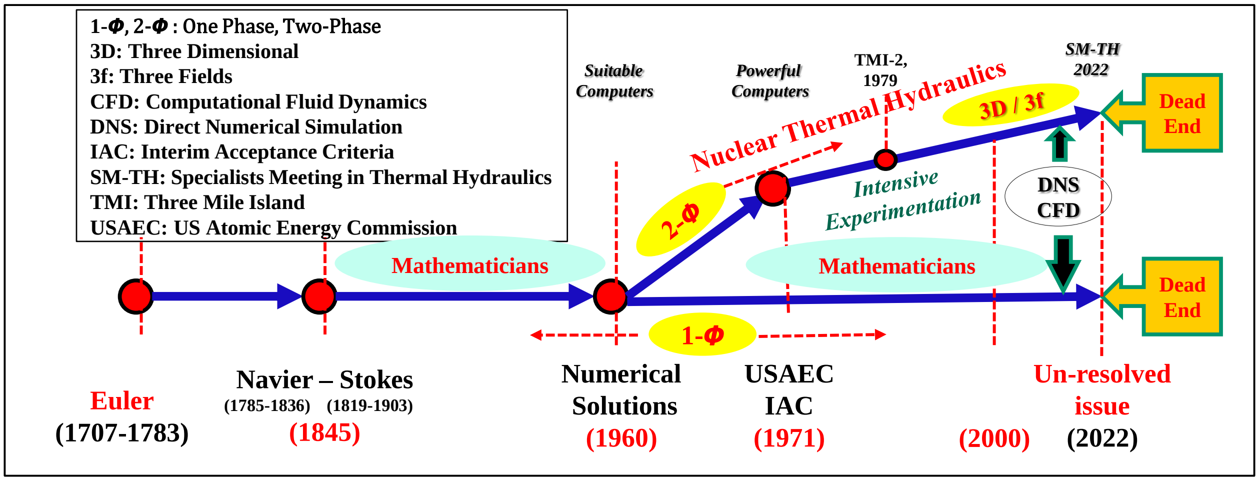

Figure 1. Navier-Stokes equations: from Euler to the un-resolved issue.

NTH, approximately in the 1960, constituted a new branch of (pseudo)-science when two-phase flow modeling became a need for demonstrating nuclear reactor safety and for progressing in the design of those reactors. Experimentation supports the NTH modeling: tailoring of models and numerical solutions to measured conditions provides credibility to the (pseudo)-science. Breakthrough events were the requirements issued by USAEC, [7], that obliged industry to an intensive research, and the Three Mile Island event [8], that moved the attention of scientists from large break Loss of Coolant Accident (LOCA) to small break LOCA with important modeling implications.

New frontier is (qualification of) 3D models including three-fields, noticeably the droplet field. DNS and CFD modeling stays in-between the pragmatic approach of NTH and the mathematician's formulation.

Averaging and turbulence, e.g. [6], constitute (some of) key issues that nowadays lead to 'dead ends' in improvement of predictive capabilities for SYS TH codes (see section 6 for more insights). Furthermore, mathematicians and NTH experts neglect (or ignore) 'intensive experimentation' and mathematical rigor, respectively. Innovative ideas for modeling approaches are mandatory for progressing.

2.2. Closure Equations

The closure equations include (suggested definition) physical and numerical equations or set-of-equations, which are part of the PDE and are functions of the unknowns. Typically, those functions have an empirical nature and do not involve time or space derivatives. Models for convection heat transfer and pressure drops are part of the set of those equations. Current issues are, see [12] and Chapters 7-9 in [1]:

- Complexity. The (simple) motion of steam bubbles in a saturated liquid determines (and is affected by) turbulence, coalescence, collapse, drag and lift forces, rotation, displaced liquid mass, heat transfer at interface, etc. Radiation (to fluid and to surrounding solid) and convection occur simultaneously with mutual effects difficult to quantify. Empirical correlations should take into account of related terms; situations that are more complex may occur in different flow conditions (e.g. reflood).

- o Repartition. Physical observation (or experiments) provide information about global terms like pressure drop and heat transfer, which need repartition in liquid, steam and interface; this implies arbitrariness, in most of situations not supported by experimental evidence.
- o Numerics. Solution of PDE implies combination of parameter ranges in a six-dimension hyperspace at each time interval; empirical correlations do not ensure 'physical solution' in each interval of the hyperspace. Numerical terms are necessary to fill gaps of information missing or inadequately interpreted from experiments: reconstructed physical reality is questionable.
- o Inadequacy of physical modeling. Two-phase pressure drops (TPPD) constitute an example of modeling proposed since more than half a century that give rise to large errors (greater than 30%) e.g. situations at low Re when stratification and/or countercurrent flow occurs.

Evidently, repartition and numerics issues become more troublesome for 3D and 3f modeling.

2.3. Special Models

Special models are necessary for predicting the thermal hydraulic evolution of nuclear reactor systems, Chapter 10 in [1]. As a difference from constitutive equations, those models need solutions at each time step outside the PDE in SYS TH codes. We distinguish two categories of special models, which aim at the simulation of:

- Components or regions of the system: centrifugal pumps and jet-pumps, steam separators and dryers, turbines and electric engines constitute typical components; the 'pre-identified' fluid mixing zones in reactor coolant system downstream the Emergency Core Coolant Injection are typical regions.
- Phenomena that are outside the boundaries of validity for the PDE including constitutive equations. Examples are the pressure drop at geometric discontinuities, the Two-Phase Critical Flow (TPCF) and the Counter Current Flow (CCF).

Errors in results can be not acceptable, e.g. in the case of TPPD at geometric discontinuity (same as friction, as discussed in section above). Available special models are empirical and provide solutions for two-phase flows typically having lower level of detail than PDE. However, CFD codes are suitable for modeling components and regions and selected phenomena, although producing errors in simulations.

3. VALIDATION

The modeling limitations brought to validation or the need of experiments to substantiate hypotheses and approximations, already in the late 70's of past century, [13], see Chapter 2 in [1]. Validation is part of Verification and Validation (V&V), where Verification is out the scope in the present paper; see Chapter 22 in [2]. Figure 2 gives a sketch of elements in validation, status and perspectives.

3.1. Experimentation

Availability of relevant experimental data is the key element to perform validation, top-left box in Fig. 2. What is the meaning of 'relevant'? Water and steam including humidity are everywhere on the earth and show-up phenomena like turbulence (e.g. around pylons of a bridge, already mentioned), condensation in open atmosphere and rains. Therefore, 'relevant' is whatever occurs in WCNR in case of nominal operation and DBA.

Here the obstacle: Owing to practical motivations (cost and safety), we cannot measure two-phase flow evolutions in reactor-transient conditions. Then we can access small size experiments that simulate or reproduce flow conditions (expected) in nuclear reactors. This gives rise to the scaling issue, section 4.

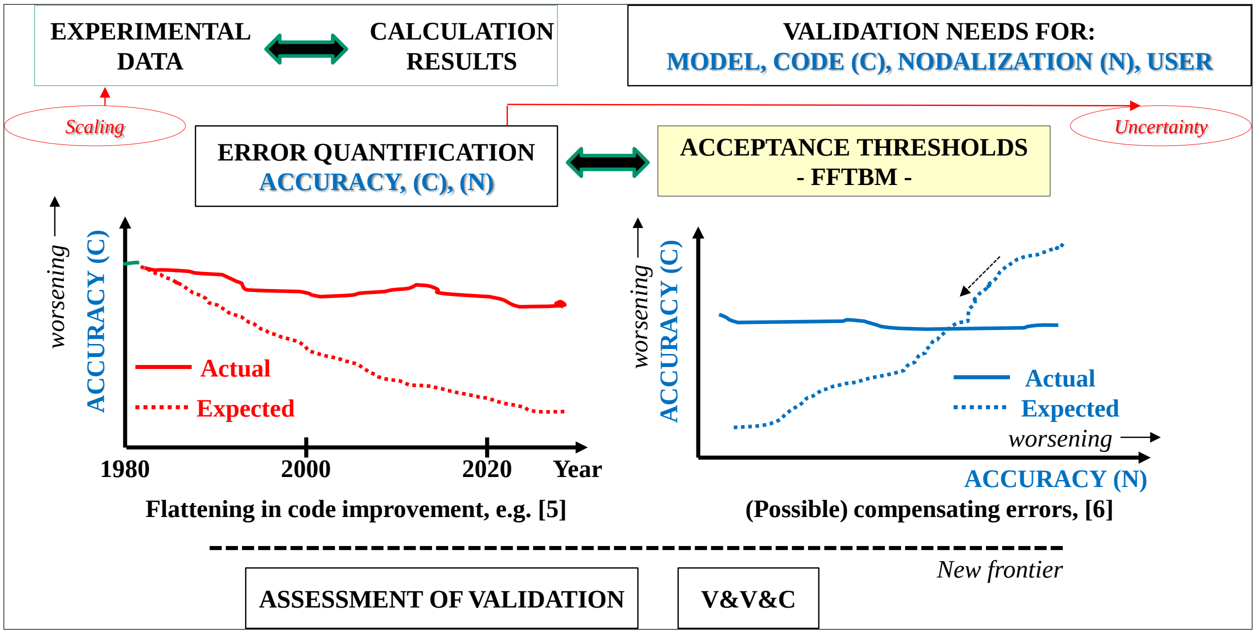

Figure 2. Elements of Validation, status and perspectives.

Design of (small-scale) experiments aims at producing data that simulate WCNR and are suitable to demonstrate the qualification level (or the capabilities) of codes. However,

- Unavoidable distortions, including dependency of data upon geometry details, prevent (in most situations) the possibility to extrapolate measured data to reactor conditions.
- Instrumentation capabilities and data qualification need improvement to demonstrate validation of recent codes, e.g. 3D, 3f, CFD, etc.

Databases of Integral Test Facilities (ITF, aimed at the simulation of the WCNR system performance) and Separate Effect Test Facilities (SETF, aimed at the simulation of components, or regions, or time slots into transients) exist in a number, looking at datasets or experiments, of several hundreds and dozens thousands, respectively.

In each experiment (or dataset), we distinguish among accident scenarios (AS), thermal hydraulic phenomena (THP) and characterizing parameters (GP), Chapter 15 in [1]. Namely, a list of 116 THP deals with all ITF and SETF experiments of interest to code validation for WCNR, [14]. Inadequate consideration by scientific community of AS and THP contributes to stagnation of the progress.

3.2. Procedures, Acceptability Thresholds and Precision Targets

Elements of the log-frame chain in-between the PDE and the calculation results related to WCNR need validation. Therefore, validation is necessary for models, codes, nodalizations and users: related procedures involve the consideration of experimental data, [15] and Chapters 13 and 22 of [2].

Procedures for accuracy quantification of code calculations by the Fast Fourier Transform Based Method [(FFTBM), (C) in Fig. 2], [16], and nodalization qualification [(N) in Fig. 2], [17], play a leading role. The sketches in the middle part of Fig. 2 provide a flash information about the application of (C) and (N) procedures:

(a) Looking at the database of thousand FFTBM applications during the last 4 decades (i.e. including analyses using experiments in the 80's), one may find a saturation for accuracy instead of a desirable decrease (left diagram).
(b) Smaller errors in setting up a nodalization (horizontal axis in right diagram) should correspond to smaller errors in calculation results (vertical axis): compensating errors prevent such a condition at least in the concerned parameter ranges, [6].

The error quantifications processes bring to the need for uncertainty, right elliptic box in Fig. 2, section 5.

Analysts easily establish acceptability thresholds for (FFTBM) accuracy prediction, i.e. based on the current practice of judging 'good' the calculation results of an assigned AS. This does not necessarily coincide with determining precision targets, which should refer to reactor safety parameters.

3.3. New Frontier

Assessment of assessment (or assessment of validation) and V&V&C (or V&V plus C = Consistency) are new frontiers for validation for SYS TH codes, bottom of Fig. 2, as discussed in Chapter 22 of [2].

The former implies the formalization of a common practice, e.g. in qualified institutions. (a) A group of qualified analysts (including code users) shall be in charge for performing meaningful validation. (b) Qualification of analysts and code users is possible following the achievement of specific training targets. (c) In addition to the application of code and nodalization validation processes, the safety relevance of validation results needs an appropriate judgment e.g. dealing with the scaling issue. .

The latter implies adding the word 'Consistency' to current V&V, [18]. A rudimentary definition follows:

'Consistency is an activity connected with the development and the qualification of numerical codes which covers topics not considered by current V&V'.

Cited references deal with motivations and features of V&V&C. Hereafter, we provide a definition with more insights into the Consistency process:

'Consistency aims at filling detected or expected gaps in the area of current V&V in NTH. Focus goes: a) to connect the modeling features and the technological needs, b) to take into account of the limitations of the experimental database mainly in terms of the space covered by parameter ranges, c) to streamlining the conditions for developing new models and to improve code capabilities'.

4. SCALING

The need for scaling derives from the small size experiments that are possible in NTH, e.g. within the validation process. Following [19], an international activity was launched that led to the Scaling State of Art Report (S-SOAR) aimed at achieving a consensus by scientific community in relation to the scaling issue, [20]. Figure 3 deals with a picture of status and interpretation of scaling based on S-SOAR.

The scaling methods, noticeably H2TS, FSA and DSS (nomenclature in the figure) are at the top of the picture considering the history and the role. Those methods primarily allow derivation of scaling factors and selected design factors from balance equations (and models), which are imperfect and constitute the motivation for scaling analyses. Furthermore, H2TS provides a summary of available knowledge about scaling in the 70's and DSS constitutes a powerful tool to perform scaling investigations, [21].

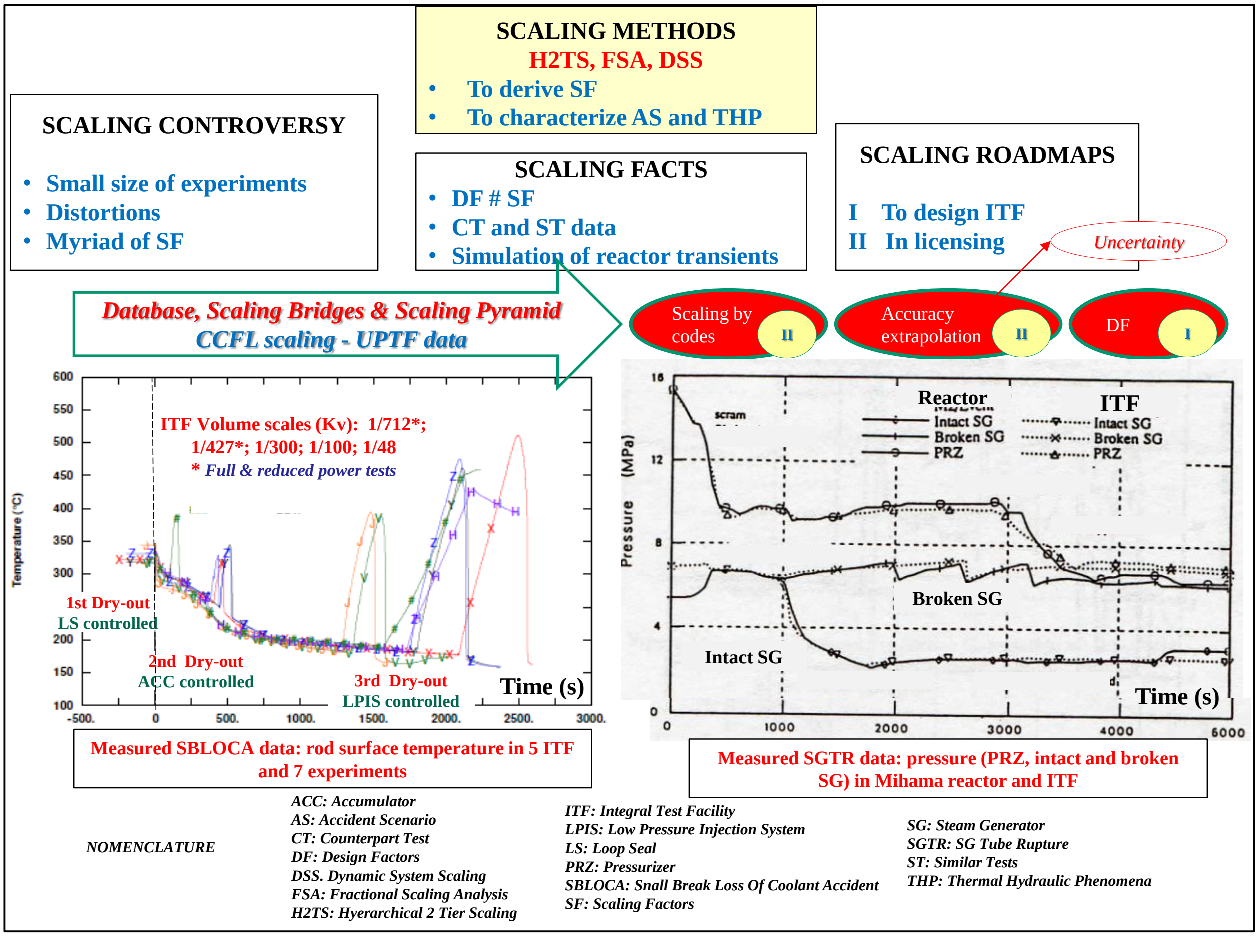

Figure 3. Elements and interpretation of scaling and scaling issue.

A controversy for the applicability and qualification of scaling analyses (top left box) characterized and still characterizes the opinion of experts. The unavoidable small size of experiments, the related scaling distortions and the myriad SF, not fulfilled at the same time, contributes to the controversy sometimes referred as the (unsolved) scaling issue.

4.1. Scaling Databases and Use

The solution to the scaling issue passes through the database of experiments, the scaling bridges and the scaling pyramid (top central box, arrow and bottom diagrams); here the prototype reactor constitutes the edge of the pyramid and the bridges connect achievements from analyses with the expected behavior of the prototype. Key sample outcomes from scaling investigations are:

- CCFL scaling confirmed, among other things, the impossibility to extrapolate experimental data to prototype conditions, [22].
- Large-scale facility data, including UPTF, [23], allowed the possibility to demonstrate code capabilities at full reactor scale.
- The analysis of seven SBLOCA Counterpart Tests (CT) performed during three decades in five differently scaled ITF, e.g. [24 - 25], leads to the following messages:
    - 5 experiments started at about 15% core power; 2 experiments started at 100% core power: all experiments showed the same THP and differences in measured parameters (time trends) are thoroughly explained.

- Main coolant pump inertia (during coast-down after trip) affect AS during initial couple of minutes: no key impact upon remaining (about one hour) transient.
- Three dry-out situations (measured/near missed) occur in all experiments: as a function of time, these are LS-controlled, ACC-controlled and LPIS-controlled.
- The same codes reproduce with acceptable accuracy (quantified by FFTBM) each experimental database.

- A nearly two-hours SGTR event (time period considered for the experimental simulation) occurred in Mihama reactor was successfully reproduced in one ITF experiment, [26 -27],
- DF allowed defining of relevant design features of ITF and boundary and initial conditions for the CT and the Mihama experiments.
- SF play a role for understanding of data, namely THP (see lists of DF and selected SF in [19]), including hierarchy of importance achievable from scaling methods. However, a myriad of possible SF, e.g. valid at each time of an AS in one region of the system, may cause counterfeiting indications for the foreseeable prototype behavior. In addition, differences in value for the same SF, e.g. derived from two scaled tests, are not justifiable without impractical analyses.

Knowledge of experimental database and results from code applications, i.e. CCFL, UPTF (and connected 2D/3D research programs [23]), CT and Mihama SGTR, Fig. 3, is essential for judging adequacy of scaling in NTH. Additional relevant scaling database includes the LOFT project, [28], and the natural circulation ST, [29], whereas ref. [20] provides a comprehensive information.

4.2. Outcomes from S-SOAR and Perspectives

The main outcome from the S-SOAR is the characterization of two scaling roadmaps, (I) and (II), having different targets, i.e. to design an ITF and to demonstrate scaling capabilities of codes in licensing (top right box in Fig. 3). Consequently (red background ellipses),

1) Based on the distinction between DF and SF, primarily DF are suitable for ITF and experiments designs, i.e. avoiding that THP which are the target of scaling investigations affect experiments designs.
2) Connected with previous item, full height scaling for ITF (or DF = 1, for elevation in gravity environment) is recommended for AS simulation in WCNR.
3) Computer codes constitute a viable and recommended option for scale-up studies, i.e. transferring experimental evidence from small scale systems to prototype (possible CT and ST) and including the demonstration of scaling capability for those codes.
4) Accuracy extrapolation, i.e. not the extrapolation of measured data, neither the extrapolation of possible code calculation results of AS at different scales, but extrapolation of error parameters involving experimental and predicted data, is a viable way for uncertainty (section 5).

A perspective in the area of scaling is to spread the consensus results from S-SOAR to the scientific community outside those who have set-up or endorsed the report. A new frontier is the demonstration of scaling capabilities of the constitutive laws, systematically including the 116 THP, [14].

5. UNCERTAINTY

The validation process, made concrete by the International Standard Problems (ISP) at CSNI during 70's and 80's, showed the impossibility to decrease the errors in code predictions below certain limits (widely different for different conditions). Thus, errors affect the prediction of AS in WCNR including for licensing purposes. The words 'uncertainty in code prediction' appeared and a research campaign started to establish uncertainty methods, Fig. 4.

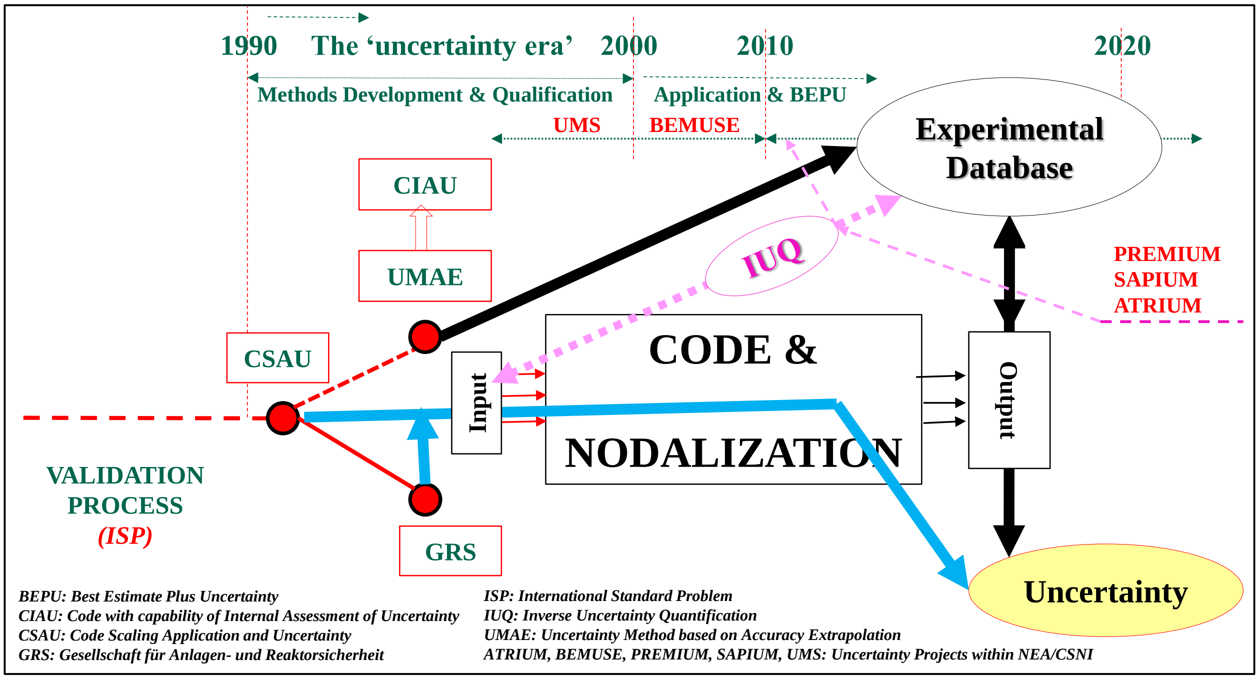

Figure 4. Uncertainty methods.

5.1. The Methods

The description of uncertainty methods is far beyond the scope here: we mention selected features, supporting Fig. 4, which are relevant for the perspective discussion.

USNRC pioneered the development of uncertainty methods with the issue of CSAU (nomenclature in Fig. 4), [30]. This is a basket of relevant requirements and ideas rather than a robust methodology. The GRS method [31] and the UMAE [32], which evolved into CIAU [33], took direct and indirect inspiration from CSAU, respectively, and brought to practical applications. Propagation of errors from selected input uncertain parameters and accuracy extrapolation are at the basis of those methods (thick blue and black arrow-lines in Fig. 4), which underwent qualification process within UMS and BEMUSE projects, Chapter 13 of [1]. Comparison of methods constitutes the topic of [34-35].

A third approach for uncertainty evaluation implies the use of powerful mathematical techniques for sensitivity analysis and the exploitation of experimental data to determine suitable matrices of variations for (all) input parameters, [36]. Nowadays, we call this IUQ, pink-dotted arrow in Fig. 4, erroneously next section) restricted to a single or a few input uncertain parameters.

5.2. The Drawbacks

Drawbacks characterize the three approaches for evaluating the uncertainty.

(1) Propagation of input uncertainties: a restricted number of input parameters ($< 10^2$, far away from the $\approx 10^5$ values needed to construct the input of a WCNR) is functional for uncertainty evaluation; related variation ranges are typically arbitrary and shall take into account of errors coming from non-considered parameters. The origins of uncertainty that derive from averaging and turbulence modeling, mentioned in section 2, are not directly part of the process. Furthermore, error propagations occur within the same (imperfect) code that is the target for uncertainty investigation.

(2) Extrapolation of output errors. The analysis of thousands experiments using the same code version and the same strategy for nodalization development (plus a number of conditions for accuracy extrapolations) needs resources not easily available; property of experimental data may constitute an unsurmountable problem. In the case of lack of experiments, e.g. future reactors, no application of the method is possible.

(3) IUQ approaches. The applicability of the method is restricted to simple situations far away from the complexity of reactors. The fashionable (due to availability of powerful mathematical techniques) IUQ processes may reveal dead-ends for progressing in the area: the association between experiments and input uncertain parameters, to determine ranges of variations, implies arbitrariness.

6. APPLICATION

The calculation processes aimed at the design and the safety of WCNR constitute the target for SYS TH code application, as shown in the central part of Fig. 5. Accident scenarios, thermal hydraulic phenomena and parameters, Chapter 15 of [1], are the outcomes of the process and lead to the characterization of safety margins. The prioritization of research in NTH is a possible additional outcome, [37].

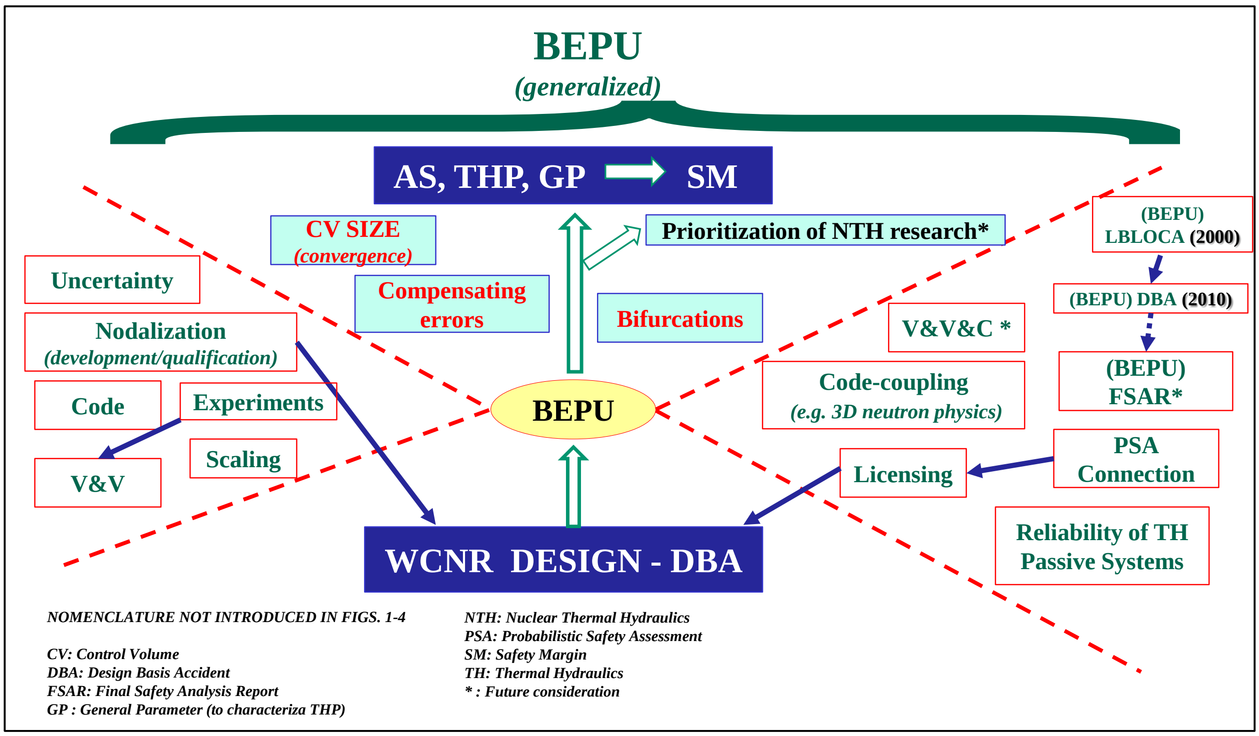

Figure 5. The code application and the proposal for generalized BEPU definition.

6.1. BEPU

The BEPU approach (middle yellow box in Fig. 5) makes possible the calculation processes. BEPU developments occurred within NTH, where (best estimate) SYS TH codes and V&V constitute the pillars. The description of BEPU, [38], brings to distinguish 'consensus' and 'proposal' elements, left and right triangular regions in Fig. 5, respectively. The proposed generalized definition (of BEPU) encompasses all the topics in Fig. 5. Sections 2-5 deal with the consensus-elements. The proposal-elements include:

- Code-couplings and licensing requirements, which are mandatory to perform safety analyses.

- PSA investigations which allow the selection of AS and of boundary conditions associated to the actuations of safety systems.
- Star-items for possible consideration in the future, in addition to the prioritization of research:
    - V&V&C outlined in section 3.3;
    - The BEPU FSAR or the third step in the evolution of BEPU, where (top right in Fig. 5), a) the first step (year 2000) is restricted to LBLOCA, [39], b) the second step (year 2010) deals with DBA transients, [40], c) the third step deals with all the parts of FSAR where analytical derivations are present, [41].

The interest towards new generation reactors suggests the application of BEPU for the reliability prediction of passive systems where thermal hydraulics is relevant in design and operation [42].

6.2. Control Volumes, Bifurcation and Compensating Errors

Control volumes are space regions of a nuclear reactor where we get solutions from PDE including closure equations and special models. The ensemble of CV (typically 10^4) constitutes the nodalization or the interface between the code and the system object of the simulation. Strictly following code manuals for set-up of CV is mandatory, as well as consideration of geometric domains for derivation of the constitutive equations and the special models in the codes, e.g. [11] and [12]. Reducing the size of CV does not bring to convergence of results; rather, an optimal nodalization exists for each code application.

Lack of convergence of results following small changes in CV size and disposition, bifurcations in transient evolutions and compensating errors, other than affecting calculation results, contribute to the disbelief in SYS TH code capabilities. Those issues, red text in upper-central part of Fig. 5, are intimately interconnected and somewhat linked with user effect and user qualification. Properly designed sensitivity studies are necessary to provide suitable quality for any complex calculation, [15] and Chapter 22 of [2].

Bifurcation analyses are possible, [43]: two identified bifurcations categories deal with occurrence of modeled cliff-edge type of phenomena (noticeably, dry-out and critical flow) and actuation of components like pumps, accumulators and valves. Uncertainty of bifurcated transients is predictable.

Compensating errors have a subtle impact upon results, as well established in NTH area, i.e. [44], [34], [6] and [45]. The constitutive equations, the averaging process, the set-up of nodalizations, the adopted values of boundary and initial conditions (sometimes controlled by user effect) combine among each other and are at the origin of compensating errors: good simulation results and wrong set of input data may co-exist.

7. CONCLUSIONS

The paper deals with an interpretation of nuclear thermal hydraulics with focus on WCNR design and transient conditions with geometrically intact core. System codes rather than CFD codes are the main concern. The related deficiencies or lacks of knowledge have no or negligible role for safety of nuclear reactors, although we provocatively started the paper in this connection.

The first conclusive remark is for non-water-cooled reactors and severe accidents thermal hydraulics technologies, i.e. outside the domain of investigation here. The huge investments in researches and the achieved sophistication level, resulting from the paper, did not prevent catastrophic events and the persisting lack of knowledge. Centuries may reveal necessary to achieve maturity in the areas of predictive capabilities for non-water-cooled reactors and severe accidents.

Code modeling inadequacies exist: turbulence modeling and averaging (or adoption of a finite control volume for the solution of the equations) are the major ones. As an answer to the question mark in the

Introduction, deeper understanding of turbulence in the river may help modeling in nuclear area. Pressure drop modeling, two-phase friction and at geometric discontinuities, is a priority candidate for future research. In this connection, a methodology for research prioritization is available. Breakthrough ideas involving long-term plans and cooperation among mathematicians, modeling and computer experts and experimentalists are necessary for a steep progress in two-phase thermal hydraulics.

Validation procedures exist, too, supported by (huge) experimental databases: those who mistrust the results of safety analysis for nuclear reactors sometimes ignore this. In this area, lack of precision targets and lack of systematic use of accuracy quantification processes (i.e. not only FFTBM), specifically true for the nodalization qualification, does not favor the progress. The proposals for V&V&C, where 'C' is consistency and assessment of assessment need resources for the actuation and eventually bring benefits.

Achieving consensus and detailed knowledge of proposed procedures and existing databases is necessary for scaling.

Drawbacks characterize uncertainty methods. Qualification of those methods, involving the use of experimental data and application of different approaches to the same problem, with consistent results (i.e. similar uncertainty bands), may satisfy technology needs, without removing drawbacks. The inverse uncertainty quantification may reveal a dead end, as far as it concerns the progress in the area: arbitrariness of connecting individual experiments, where unavoidably many parameters and phenomena play a role, with selected input uncertain parameters, is questionable.

Compensating errors and bifurcations, together with user effect, including nodalization choices by the user, unavoidably affect the results of code calculations. Performing proper sensitivity studies and the use of methods to identify bifurcations are necessary to achieve qualified calculations.

REFERENCES

1. F. D'Auria [Ed.] (Authors: Aksan N., Bestion D., D'Auria F., Galassi G.M., Glaeser H., Hassan Y., Jeong J. J., Kirillov P., Morel C., Ninokata H., Reventos F., Rohatgi U., Schultz R. R., Umminger K.), *Thermal Hydraulics in Water-Cooled Nuclear Reactors*, Elsevier, Woodhead Publishing, Duxford UK pp 1-1221 (2017).
2. F. D'Auria, and Y. Hassan [Ed.s], *Handbook of Thermal Hydraulics in Water-Cooled Nuclear Reactors* (or, 2nd Edition of book at [1]), Elsevier, Woodhead Publishing, Duxford UK (2023), to be issued
3. OECD/NEA/CSNI, *Summary Report (F. D'Auria Ed.) in Proc. of CSNI Specialists Meet. on Transient Thermal-hydraulics in Water Cooled Nuclear Reactors (SM-TH)*, Madrid, Spain, March 22-24, NEA Publisher, Paris France (2023), to be issued
4. R.B. Duffey and F. D'Auria, "Nuclear Energy and its History: Past Consequences, Present Inadequacies and a Perspective for Success", *Energy and Power Engineering*, **12**, pp 193-236 (2020)
5. N. Aksan and F. D'Auria, "Summary from 'Aix-en-Provence 1992' and 'Annapolis 1996' OECD/NEA/CSNI Specialists Meetings in Nuclear Thermal Hydraulics", *Proc. of CSNI Specialists Meet. on Transient Thermal-hydraulics in Water Cooled Nuclear Reactors (SM-TH),* Madrid, Spain, March 22-24, Vol I, p1, pp 1-20 (2022).
6. F. D'Auria and Y. Hassan, "Challenges and concerns for development of nuclear thermal-hydraulics", *Nuclear Engineering and Design*, **375** (111074), pp 1-12 (2021).
7. USAEC, "Interim Acceptance Criteria (IAC) for ECCS". Washington, DC, USA (1971).
8. R.E Henry, *TMI-2: An Event in Accident Management for Light-Water-Moderated Reactors*. ANS, LaGrange Park USA (2011).
9. L.C. Berselli, *Three-Dimensional Navier-Stokes Equations for Turbulence*, Elsevier, Academic Press Publisher, Duxford (UK), pp 1-328 (2021).

10.J.B. Joshi and A.K. Nayak [Ed.s], *Advances of Computational Fluid Dynamics in Nuclear Reactor Design and Safety Assessment*, Elsevier, Woodhead Publishing, Duxford (UK), pp 1-850, (2017).
11.N. Dinh, R. Nourgaliev, A. Bui and H. Lee, "Perspectives on Nuclear Reactor Thermal Hydraulics", *Proc. of the International Top. Meet. on Nuclear Reactor Thermal Hydraulics (NURETH-15),* Pisa Italy, May 12-17, (2013).
12.M.Z. Podowski, "Understanding two-phase flow and boiling heat transfer: challenges and paradoxes", *Nuclear Engineering and Design [Special Issue devoted to B.R. Sehgal, G. Yadigaroglu and G.F. Hewitt, Ed.s F. D'Auria and Y. Hassan]*, **354** (110185), pp 1-10 (2019).
13.OECD/NEA/CSNI, [Authors: Aksan N., Bessette D., Brittain I., D'Auria F., Gruber P., Holmstrom H.L.O., Landry R., Naff S., Pochard R., Preusche G., Reocreux M., Sandervag O., Staedtke H., Wolfert K., Zuber N.], *CSNI Code Validation Matrix of Thermo-Hydraulic Codes for LWR LOCA and Transients*, CSNI, 132, Paris France, (1987).
14.N. Aksan, F. D'Auria and H. Glaeser, "Thermal-hydraulic phenomena for water cooled nuclear reactors", *Nuclear Engineering and Design*, **330**, pp 266-286 (2018).
15.A. Petruzzi and F. D'Auria, "Standardized Consolidated Calculated and Reference Experimental Database (SCCRED): a Supporting Tool for V&V and Uncertainty Evaluation of Best-Estimate System Codes for Licensing Applications", *Nuclear Science and Engineering*, **182** (1), pp 13-53 (2016).
16.W. Ambrosini R. Bovalini and F. D'Auria, "Evaluation of Accuracy of Thermal-hydraulic Codes Calculations", *Energia Nucleare*, **7** (2), (1990).
17.M. Bonuccelli, F. D'Auria, N. Debrecin and G.M. Galassi, "A methodology for the qualification of thermal–hydraulic codes nodalizations", *Proc. of the International Top. Meet. on Nuclear Reactor Thermal Hydraulics (NURETH-6)*, Grenoble, France (1993).
18.F. D'Auria and M. Lanfredini, "V&V&C in nuclear reactor thermal-hydraulics", *Nuclear Engineering and Design [Special Issue devoted to B.R. Sehgal, G. Yadigaroglu and G.F. Hewitt, Ed.s F. D'Auria and Y. Hassan]*, **354** (110162), pp 1-10 (2019).
19.F. D'Auria and G. Galassi, "Scaling in nuclear reactor system thermal-hydraulics", *Nuclear Engineering and Design*, **240**, pp 3267-3293 (2010).
20.OECD/NEA/CSNI, [(Lead Authors) Bestion D., D'Auria F. (Ed.), Lien P., Nakamura H.], "*Scaling in system thermal-Hydraulics applications to Nuclear Reactor safety and design: a State-of-the-Art-Report*", NEA/ CSNI/R(2016)14, JT03411050, Paris, France, pp 1-393, (2017).
21.J.N. Reyes Jr., "The Dynamical System Scaling Methodology", *Proc. of the International Top. Meet. on Nuclear Reactor Thermal Hydraulics (NURETH-16)*, Chicago, Ill, USA (2015).
22.H. Glaeser, "Downcomer and tie plate countercurrent flow in the Upper Plenum Test Facility (UPTF)", *Nuclear Engineering and Design*, **133**, pp 259-283 (1992).
23.P.S. Damerell and J.W. Simons, "Reactor Safety Issues Resolved by the 2D/3-D Program", USNRC, NUREG/IA-0127 (USA), GRS-101 (G), MPR—1346 (US), JAERI 1336 (J), Washington D.C. (1993).
24.A. Del Nevo, F. D'Auria, M. Mazzini, M. Bykov, I. Elkin and A. Suslov, "The Design of PSB-VVER Experiments Relevant to Accident Management", *Power and Energy Systems*, **2** (1), pp 371-385, (2008).
25.B. Mavko, A. Prosek and F. D'Auria, "Determination of code accuracy in predicting small break LOCA experiment", *Nuclear Technology*, **120**, pp 1-18 (1997).
26.M. Hirano and T. Watanabe, "Analysis of the Mihama-2 SGTR Event and ROSA-IV Experiment", *Proc. of the International Top. Meet. on Nuclear Reactor Thermal Hydraulics (NURETH-5)*, Salt Lake City, UT USA, Sept 21-4, pp 165-173 (1992).
27.S. Yokobori and H. Nakamura, "Optimized Apportion between Large-Scale Nuclear Thermal-hydraulic Tests and Simulation", *Japan Atomic Energy*, **27**, pp 461-468 (2008).
28.S,M. Modro, L.D. Goodrich and G.D. McPherson, "Scaling and instrumentation of the LOFT facility", *Proc. of Specialist Meet. on Small Break LOCA Analysis in LWRs*, Pisa, Italy (1985).
29.F. D'Auria and M. Frogheri, "Use of Natural circulation map for assessing PWR performance", *Nuclear Engineering and Design*, **215** (1/2), pp 111-126 (2002).

30. N. Zuber, G.E. Wilson, B.E. Boyack, I. Catton I., R,B. Duffey, P., Griffith, K.R. Katsma, G.S. Lellouche, S., Levy, U.S. Rohatgi and W. Wulff, "Quantifying reactor safety margins, part 5: evaluation of scale-up capabilities of best estimate codes", *Nuclear Engineering and Design*, **119**, pp 97-107 (1990).
31. H. Glaeser, "Uncertainty evaluation of thermal-hydraulic code results", *Proc. of Best Estimate Methods in Nuclear Installations Safety Analysis*, Washington, DC USA (2000).
32. F. D'Auria, N. Debrecin and G.M. Galassi, "Outline of the uncertainty methodology based on accuracy extrapolation", *Nuclear Technology,* **109** (1), pp 21–38 (1995).
33. F. D'Auria and W. Giannotti, "Development of code with capability of internal assessment of uncertainty", *Nuclear Technology*, **131** (1), pp 159–196 (2000).
34. A. J. Wickett and G. Yadigaroglu, "Report of a CSNI Workshop on Uncertainty Analysis Methods", *Proc. of a Workshop held in London on March 1-4 1994*, NEA/CSNI/R(94)20/Part l, Paris France, pp 1-247 (1994).
35. F. D'Auria, H. Glaeser, S. Lee, J. Mišák, M. Modro and R.R. Schultz,"Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation", *IAEA Safety Report Series, SRS No 52*, Vienna Austria, pp 1-211 (2008).
36. A. Petruzzi, D.G. Cacuci and F. D'Auria, "Best-Estimate Model Calibration and Prediction through Experimental Data Assimilation - II: Application to a Blow-down Benchmark Experiment", *Nuclear Science and Engineering,* **165**, pp 1-52 (2010).
37. F. D'Auria, "Prioritization of nuclear thermal-hydraulics researches", *Nuclear Engineering and Design*, **340**, pp 105-111 (2018).
38. F. D'Auria, "Best Estimate Plus Uncertainty (BEPU): Status and perspective", *Nuclear Engineering and Design*, **352**, 110190, pp 1-11 (2019)
39. M.R. Galetti and F. D'Auria, "Questions arising from the application of Best-Estimate Methods to the Angra-2 NPP Licensing Process in Brazil", *Int. Meet. on Best-Estimate Methods in Nuclear Installation Safety Analysis*, Washington D.C., USA (2000).
40. F. D'Auria, G.M. Galassi, O. Mazzantini and J. Riznic [Ed.], *Pressurized Heavy Water Reactors – Atucha II, Vol 8 (Series Editor, Y. Koizumi)*, Elsevier, Netherlands, pp 1-587 (2021).
41. F. Menzel, G. Sabundijan, F. D'Auria and A. Madeira, "Proposal for systematic application of BEPU in the licensing process of nuclear power plants", *Nuclear Energy Science and Technology*, **10** (4), pp 323-338 (2016).
42. F. D'Auria, "Passive systems and nuclear thermal hydraulics", *Nuclear Engineering and Design*, **385**, 111513, pp 1-7 (2021).
43. F. D'Auria, W. Giannotti and A. Piagentini, "Consideration of Bifurcations within the Internal Assessment of Uncertainty", *(invited at) ASME-JSME Int. Conf. on Nuclear Engineering (ICONE-8)*, Baltimore, Md USA (2000).
44. W. Ambrosini, F. D'Auria, W. Grassi and P. Vigni, "Accuracy in the Prediction of Two-Phase Flow Regimes", *Proc. of Eurotherm Sem. No. 3: Modeling of Nuclear and Advanced Heat Transfer Components*, Bologna (I), June 14-15 (1988).
45. D. Bestion, F. D'Auria and M. Lanfredini, "Compensation of errors in reactor numerical and experimental simulations", *Proc. of the International Top. Meet. on Nuclear Reactor Thermal Hydraulics (NURETH-19),* Brussels Belgium, March 6-11 (2022).