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Review Article

Thermal-Hydraulic System Codes in Nulcear Reactor Safety and Qualification Procedures

Alessandro Petruzzi and Francesco D'Auria

DIMNP, University of Pisa, Via Diotisalvi 2, 56100 Pisa, Italy

Correspondence should be addressed to Alessandro Petruzzi, a.petruzzi@ing.unipi.it

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In the last four decades, large efforts have been undertaken to provide reliable thermal-hydraulic system codes for the analyses of transients and accidents in nuclear power plants. Whereas the first system codes, developed at the beginning of the 1970s, utilized the homogenous equilibrium model with three balance equations to describe the two-phase flow, nowadays the more advanced system codes are based on the so-called "two-fluid model" with separation of the water and vapor phases, resulting in systems with at least six balance equations. The wide experimental campaign, constituted by the integral and separate effect tests, conducted under the umbrella of the OECD/CSNI was at the basis of the development and validation of the thermal-hydraulic system codes by which they have reached the present high degree of maturity. However, notwithstanding the huge amounts of financial and human resources invested, the results predicted by the code are still affected by errors whose origins can be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, and imperfect knowledge of boundary and initial conditions. In this context, the existence of qualified procedures for a consistent application of qualified thermal-hydraulic system code is necessary and implies the drawing up of specific criteria through which the code-user, the nodalization, and finally the transient results are qualified.

1. INTRODUCTION

Evaluation of nuclear power plants (NPPs) performances during accident conditions has been the main issue of the research in nuclear fields during the last 40 years. Therefore, several complex system thermal-hydraulic codes have been developed for simulating the transient behavior of water-cooled reactors. In the early stage of the development, the codes were primarily applied for the design of the engineered safety systems. In 1978, the "appendix K requirements" [1] were issued, defining conservative model assumptions as well as conservative initial and boundary conditions to warrant conservative code results for critical safety parameters. On the other hand, the development and elaboration of accident management procedures, the application of probabilistic safety analyses (PSA) and the operator training asked for so-called "best-estimate (BE) analysis," that means an accident simulation as realistic as possible. The main objective of best-estimate system codes was to replace the "evaluation models," which used many conservative assumptions, by the best-estimate approach for more realistic predictions of pressurized water reactor (PWR) or boiling water reactor (BWR) accidental transients that allow the reduction of safety margins. Best-estimate system codes are currently used for the following:

(i) safety analysis of accident scenarios;
(ii) quantification of the conservative analyses margin;
(iii) licensing purposes if the code is used together with a methodology to evaluate uncertainties;
(iv) probabilistic safety analysis (PSA);
(v) development and verification of accident management procedures;
(vi) reactors design;
(vii) analysis of operational events;
(viii) core management investigation.

Best-estimate thermal-hydraulic codes (e.g., RELAP, TRAC, CATHARE, ATHLET, ...) are, in general, based on equations for two-phase flow which are typically resolved in Eulerian coordinates. The two-phase flow field is described by mass, momentum, and energy conservation equations for the liquid and vapour phases separately and mass conservation equations for noncondensable gas present in the mixture. The models are suitable for 1D system simulation even if for some NPP component (e.g., the vessel), some code has the capability to solve 3D system equations. Time discretization could be fully, semi or nearly implicit. Depending on the number of balance equations, different sets of constitutive equations are required to close the equation system. In comparison with the homogeneous equilibrium model (HEM), which requires only two constitutive equations, namely, the friction loss and the heat transfer relations at the wall, at least seven constitutive equations are required for the two fluid models with six balance equations describing the mass, energy, and momentum transfers at the interface and the energy and momentum transfers of the water- and steam-phase at the wall. The constitutive equations have to describe the physical phenomena in a wide span of scale, ranging from down-scaled integral system experiments up to full size reactor geometry. This is one of the most challenging goals in code development and code validation. To develop and validate the scaling laws for individual phenomena, separate effect tests in different scale are necessary. In Figure 1, the code development activities carried out in more than three decades are shown.

Due to the numerical approximations and the empirical nature of the included models in the thermal-hydraulic system codes, extensive activities related to validation of the codes have been pursued during the years. The validation has been performed using experimental data from specially designed scaled-down test facilities. In addition, transient data from real NPPs were also considered due to the full scale and true geometry although those data concern only conditions under fairly mild transients (operational transients and start-up and commissioning tests). These activities have been planned and carried out in national and international contexts in four levels, mainly in the independent assessment area, involving the use of the following:

(a) "fundamental" experiments [2];
(b) separate effects test facilities (SETF) [3];
(c) integral test facilities (ITF), including most of the international standard problems (ISP) [4];
(d) real plant data.

However, notwithstanding the huge amounts of financial and human resources invested, the results predicted by the code are still affected by errors whose origins can be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, and imperfect knowledge of boundary and initial conditions. In this context, the existence of qualified procedures for a consistent application of qualified thermal-hydraulic system code is necessary and implies the drawing up of specific criteria through which the code-user, the nodalization, and finally the transient results are qualified.

The current situation related to the development, validation, and use of system codes can be summarized as follows.

(1) A state-of-the-art report in modeling LOCA (loss-of-coolant accident) and non-LOCA transients and the compendium on ECCS (emergency core cooling systems). Researches have been published in 1989 [5, 6], by Organization for Cooperation and Development/Committee on the Safety of Nuclear Installations (OECD/CSNI) and US NRC. These reports broadly cover topics like plant features relevant to thermal-hydraulics, transient description, phenomena identification, code modeling capabilities and needs for experimental data and present situation in the experimental area.
(2) The CSAU (Code Scaling, Applicability and Uncertainty), published in 1990, for example [7], constituted a pioneering effort made by NRC in the area of code uncertainty prediction.
(3) Code validation criteria and detailed qualification programs exist, although not fully optimized or internationally agreed on. In particular, the following hold.
    (a) The integral test facility CSNI code validation matrix (ITF-CCVM) report was initially published in 1987 and extensively updated in 1996, [4]. Tests for code validation were selected based on quality of the data, variety of scaling and geometry, and appropriateness of the range of covered conditions. The decision was taken around 1984 to bias the validation matrix toward integral tests so that code models were exercised and interacted in situations as similar as possible to those of interest to PWR and BWR. This was done because of the assumption that sufficient comparison with separate effects test data would be performed and documented by code developers.
    (b) As the last expectation has proved unrealistic, a group of scientists was formed toward the end of the 80s to set up the separate effect test facility CSNI code validation matrix, SETF-CCVM, that was issued in 1994 [3]. The development of the SETF-CCVM required an extension of the methodology employed for the ITF-CCVM [4], both in the scope and the definition of the thermal-hydraulic phenomena and in the categorization and description of facilities. A significant result of the activity was the selection of sixty-seven phenomena assumed to cover all the thermal-hydraulic situations of interest expected in PWR and BWR transients. The needed effort suitable for a comprehensive code validation was quantified: more than one thousand experiments should be part of a thermal-hydraulic system code validation program. The impact of those findings in planning new researches was also evaluated [8].
(4) The codes have reached an acceptable degree of maturity although the reliable application is still limited to the validation domain.

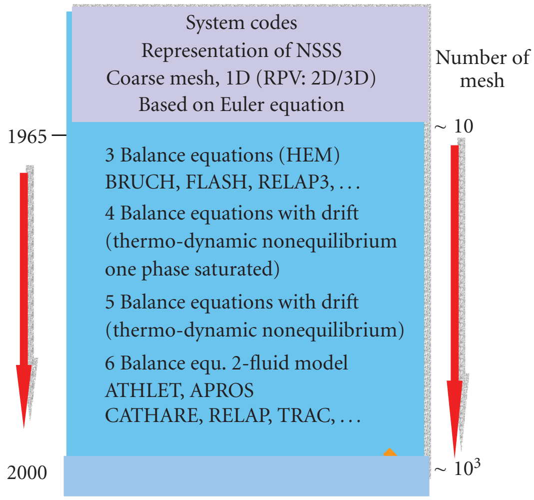

Figure 1: Code development activities in more than three decades.

(5) The use of qualified codes is more and more requested for assessing the safety of existing reactors, especially in the former Soviet Union and in the Eastern countries, and for designing advanced reactors.
(6) The codes availability is increasingly growing especially in the countries belonging to the former Soviet Union, the Eastern countries, Korea, China, and so forth.
(7) Special topics, like user [9] and computer-compiler effects upon code calculation results, nodalization qualification [10], accuracy quantification [11], relevance of international standard problems and lesson learned, use of best estimate codes in the licensing, have been widely discussed and main achievements are available to the international community.
(8) A special attention from the scientific community has always been given to the quantification of code uncertainty in predicting plant transients. Methodologies to evaluate the "uncertainty" have been proposed [12, 13] and tested in several international activities, like UMS (uncertainty method study, [14]) and BEMUSE (best-estimate methods–uncertainty and sensitivity evaluation, [15, 16]) that allowed the comparison of uncertainty results obtained from different methodologies.

This paper reviews the main features and limitations of the thermal-hydraulic system codes and the procedures adopted for the qualification of computational tools, that is, not only the codes, through the ITF and SETF validation matrixes, but also the nodalization used to simulate the transient scenario in the NPP. Finally, taking into account the multidisciplinary nature of reactor transients and accidents (which include thermal-hydraulics, neutronics, structural, and radiological aspects), the needs, the status of development, and the benefits of code coupling are pointed out.

2. MAIN FEATURES AND LIMITATIONS OF THERMAL-HYDRAULIC SYSTEM CODES

The system thermal-hydraulic codes are based upon the solution of six balance equations for liquid and steam that are supplemented by a suitable set of constitutive equations. The balance equations are coupled with conduction heat transfer equations and with neutron kinetics equations (typically point kinetics). The two-phase flow field is organized in a number of lumped volumes connected with junctions. Thermal-hydraulic components such as valves, pumps, separators, annulus, accumulators, and so forth, can be defined in order to represent the overall system configuration. In the following sections, main problematic aspects, from the point of view of the user, of a thermal-hydraulic system code are highlighted.

2.1. System nodalization

All major existing light water reactor (LWR) safety thermal-hydraulics system codes follow the concept of a "free nodalization," that is, the code user has to build up a detailed noding diagram which maps the whole system to be calculated into the frame of a one-dimensional thermal-hydraulic network. To do this, the codes offer a number of basic elements like single volumes, pipes, branches, junctions, heat structures, and so forth. This approach provides not only a large flexibility with respect to different reactor designs, but also allows predicting separate effect and integral test facilities which might deviate considerably from the full-size reactor.

As a consequence of this rather "open strategy," a large responsibility is passed to the user of the code in order to develop an adequate nodalization scheme which makes best use of the various modules and the prediction capabilities of the specific code. Due to the existing code limitations and to economic constraints, the development of such a nodalization represents always a compromise between the desired degree of resolution and an acceptable computational effort. It is not possible here to cover all the aspects of the development of an adequate nodalization diagram, however, two crucial problems will be briefly mentioned which illustrate the basic problem.

2.1.1. Spatial convergence

As has been quite often misunderstood, a continuous refinement of the spatial resolution (e.g., a reduction of the cell sizes) does not automatically improve the accuracy of the prediction. There are two major reasons for this behavior:

(1) the large number of empirical constitutive relations used in the codes has been developed on the basis of a fixed (in general coarse) nodalization;
(2) the numerical schemes used in the codes generally include a sufficient amount of artificial viscosity which is needed in order to provide stable numerical results. A reduction of the cell sizes below a certain threshold value might result in severe nonphysical instabilities.

From those considerations, it can be concluded that no a priori optimal approach for the nodalization scheme exists.

2.1.2. Mapping of multidimensional effects

Multidimensional effects, especially with respect to flow splitting and flow merging processes (e.g., the connection of the main coolant pipe to the pressure vessel), exist also in relatively small scale integral test facilities. The problem might become even more complicated due to the presence of additional bypass flows and a large redistribution of flow during the transient. It is left to the code user to determine how to map these flow conditions within the frame of a one-dimensional code, using the existing elements like branch components, multiple junction connections, or cross-flow junctions. These two examples show how the limitations in the physical modeling and the numerical method in the codes have to be compensated by an “engineering judgment” of the code user which, at best, is based on results of detailed sensitivity of assessment studies. However, in many cases, due to lack of time or lack of appropriate experimental data, the user is forced to make ad hoc decisions.

2.2. Code options: physical model parameters

Even though the number of user options has been largely reduced in the advanced codes, various possibilities exist about how the code can physically model specific phenomena. Some examples are as follows.

(1) Choice between engineering type models for choking or use of code implicit calculation of critical two-phase flow conditions.
(2) Flow multipliers for subcooled or saturated choked flow.
(3) The efficiency of separators.
(4) Two-phase flow characteristics of the main coolant pumps.
(5) Pressure loss coefficient for pipes, pipe connections, valves, branches, and so forth.

Since in many cases direct measured data are not available or, at least, not complete, the user is left to his engineering judgment to specify those parameters.

2.3. Input parameter related to specific system characteristics

The assessment of LWR safety codes is mainly performed on the basis of experimental data coming from scaled integral or separate effect test facilities. Typically in these scaled-down facilities, specific effects, which might be small or even negligible for the full-size reactor case, can become as important as the major phenomena to be investigated. Examples are the release of the heat from the structures to the coolant, heat losses to the environment, or small bypass flows. Often, the quality of the prediction depends largely on the correct description of those effects which needs a very detailed representation of the structural materials and a good approximation of the local distribution of the heat losses. However, many times the importance of those effects is largely underestimated, and consequently, wrong conclusions are drawn from results based on incomplete representation of a small-scale test facility.

2.4. Input parameters needed for specific system components

The general thermal-hydraulic system behavior is described in the codes by the major code modules based on a one-dimensional formulation of the mass, momentum, and energy equations for the separated phases. However, for a number of system components, this approach is not adequate and consequently additional, mainly empirical models have to be introduced, for example, for pumps, valves, separators, and so forth. In general, these models require a large amount of additional code input data, which are often not known since they are largely scaling dependent.

A typical example is the input data needed for the homologous curves which describe the pump behavior under single and two-phase flow conditions which in general are known only for a few small-scale pumps. In all these cases, the code user has to extrapolate from existing data obtained for different designs and scaling factors which introduces a further uncertainty to the prediction.

2.5. Specification of initial and boundary conditions

Most of the existing codes do not provide a steady-state option. In these cases pseudo-steady-state runs have to be performed using more or less artificial control systems in order to drive the code towards the specified initial conditions. The specification of stable initial and boundary conditions and the setting of related controllers require great care and detailed checking. If this is not done correctly, a large risk, that even small imbalances in the initial data will overwrite the following transient, exists especially for slow transients and small break LOCA calculations.

2.6. Specification of state and transport property data

The calculation of state and transport properties is usually done implicitly by the code. However, in some cases, for example, in RELAP5, the code user can define the range of reference points for property tables, and therefore, can influence the accuracy of the prediction. This might be of importance especially in more “difficult regions,” for example, close to the critical point or at conditions near atmospheric pressure. Another example is constituted by the fuel materials property data: the specification of fuel rod gap conductance (and thickness) is an important parameter, affecting core dryout and rewet occurrences that must be selected by the user.

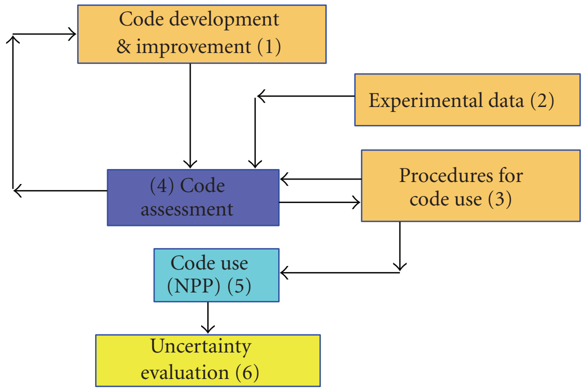

Figure 2: A consistent application (development, qualification and application) of a thermalhydraulic system code.

2.7. Selection of parameters determining time step sizes

All the existing codes are using automatic procedures for the selection of time step sizes in order to provide convergence and accuracy of the prediction. Experience shows, however, that these procedures do not always guarantee stable numerical results, and therefore, the user might often force the code to take very small time steps in order to pass through trouble spots. In some cases, if this action is not taken, very large numerical errors can be introduced in the evolution of any transient scenario and are not always checked by the code user.

2.8. Code input errors

In order to prepare a complete input data deck for a large system, the code user has to provide a huge number of parameters (approx., 15 to 20 thousand values for an NPP nodalization) which he has to type one by one. Even if all the codes provided consistency checks, the probability for code input errors is relatively high and can be reduced only by extreme care following clear quality assurance guidelines.

3. QUALIFICATION OF COMPUTATIONAL TOOLS

A key feature of the activities performed in nuclear reactor safety technology is constituted by the necessity to demonstrate the qualification level of each tool adopted within an assigned process and of each step of the concerned process. Computational tools include (numerical) codes, nodalizations, and procedures. Furthermore, the users of those computational tools are part of the process and need suitable demonstration of qualification.

A consistent application (development, qualification, and application) of a thermal-hydraulic system code is depicted in Figure 2. The code development and improvement process, block 1 in Figure 2, is conducted by "code developers" who make extensive use of assessment (block 4), typically performed by independent users of the code (i.e., group pf experts independent from those who developed the code). The consistent code assessment process implies the availability of experimental data and of robust procedures for the use of the codes, blocks 2 and 3, respectively. Once the process identified by blocks 1 and 4 is completed, a qualified code is available to the technical community, ready to be used for NPP applications (block 5). The NPP applications still require "consistent" procedures (block 3) for a qualified use of the code. The results from the calculations are, whatever the qualification level achieved by the code is, affected by errors that must be quantified through appropriate uncertainty evaluation methodology (block 6).

3.1. Code qualification

The code constitutes the main tool for investigating the NPP behavior or for evaluating the efficacy of systems or special procedures during accident transient scenarios. The following constitutes the main requisites for a qualified use of the code [11].

(1) Capability of the code to reproduce the relevant phenomena occurring for the selected spectrum of accidents.
(2) Capability to reproduce the peculiarities of the reference plant/facility.
(3) Capability to produce suitable results for a comparison with the acceptable criteria.
(4) Availability of qualified users.

Essentially the code must be able to reproduce two fundamental aspects [17].

(a) The NPP and the accident conditions: all the relevant zones, systems, procedure, and related actuation logic is to be included in the calculation. This item also includes any external event, boundary and initial condition necessary to identify the plant but also the selected accident.
(b) The phenomena occurring (expected) during the accident.

In order to ensure those capabilities, the code qualification process is needed and the following two phases can be identified.

(1) Development phase: several models are created, developed, and improved by the code development team; many checks are necessary to qualify each model and the global architecture of the code.
(2) Independent assessment phase: the code is ready to be used but qualified calculations performed by organizations independent from the code-development team are needed to check independently the declared capabilities of the code.

It is relevant to note that in the development phase the code models can be changed and the code is not available to the final user. In the independent assessment phase, the final version of the code is distributed and the user is generally forbidden to change any element of the code models apart from the normal available options as described in the user manual.

The activities performed during the development phase are (Figure 3) as follows.

(a) *Verification:* it consists in the review of the source coding relative to its description in the documentation. In other words, *code verification* involves activities that are related to software quality assurance (SQA) practices and to activities directed toward finding and removing deficiencies in models and in numerical algorithms used to solve partial differential equations. SQA procedures are needed during software development and modification, as well as during production computing. SQA procedures are well developed in general, but areas of improvement are needed with regard to software operating on massively parallel computer systems. During the verification step, the correct working of models, interfaces, and numerics is checked to ensure that the code, in all its components, is free of errors and produces results.

(b) *Validation (or assessment):* it consists in evaluating the accuracy of the values predicted by the code-nodalization against *relevant experimental* data for important phenomena expected to occur. In other words, *code validation* emphasizes the quantitative assessment of computational model accuracy by comparison with high-quality validation experiments, that is, experiments that are well characterized in terms of measurement and documentation of all the input quantities needed for the computational model, as well as carefully estimated and documented experimental measurement uncertainty. The validation process ensures the consistency of the results produced by the code; that is, it proves that the code, as a whole system, is capable to produce meaningful results: not only the code-system works, but it also works in the right direction.

The *independent code-assessment* is carried out by independent users of the code and has the aim to quantify the code accuracy, which is the discrepancy between transient calculations and *experiments performed* in ITF. The independent assessment of the code involves different aspects, like (Figure 3)

(1) qualification of the nodalization;
(2) qualification of the user;
(3) definitions of procedures for the use of the code;
(4) evaluation of the accuracy from a qualitative and quantitative point of view.

The above items are connected with the application of the code to experimental tests performed in ITF. The procedure for the qualification of the nodalization is described with more details in the Section 3.4 together with acceptability criteria.

Besides the demonstration of the code capability in reproducing an experiment performed in a test facility, the code must be checked also in performing NPP calculation. This constitutes the final step of the independent code assessment (Figure 4): the demonstration of the code capability at a different scale, that is, the full scale of the NPP. A nodalization of an NPP is prepared and qualified. The check consists in a "similarity analysis" generally involving a Kv-scaled calculation (see Section 3.3). In this kind of calculation, the initial and boundary conditions of an experiment performed in an ITF are properly scaled and implemented in the NPP nodalization. The results of the NPP-scaled nodalization must reproduce the relevant phenomena occurring in the experiment. Alternative ways to prove the code capability at the NPP scale are constituted by the comparison with other qualified NPP code results or, if available, with data obtained in NPP operational transients. As the procedure followed for this part of the code assessment is the same adopted for the qualification process of the nodalization, more details are given in Section 3.4.

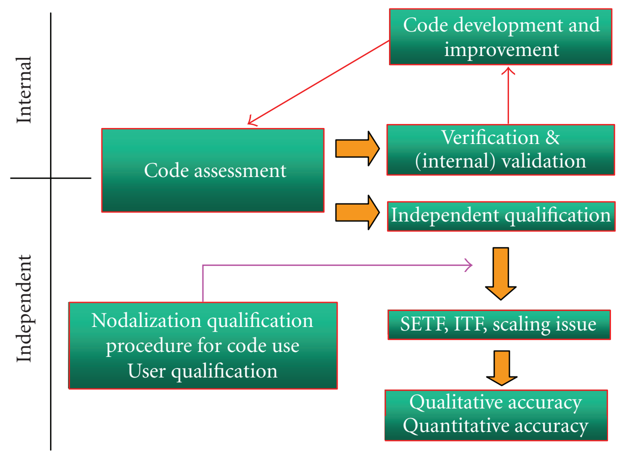

Figure 3: Internal and external (independent) code assessment.

The contemporaneous acceptability of the accuracy (step of the process connected with experiments in ITF) and of the similarity analysis (step of the process connected with NPP) constitutes the positive demonstration of the code capability and the end of the code assessment. The calculated accuracy is possibly included in the data base suitable for uncertainty evaluation (block 6 in Figure 2, [12, 13]). If the accuracy is not in the range of acceptability or the code fails the similarity analysis, the code is considered not qualified and the code-development team will be informed in order to develop new code models or to improve the existing ones.

As consequence, new revision or new version of the code can be produced during the development phase: a new revision contains a new physical modeling whereas a new version may contain new numerical methods, new modules, new submodules, new preprocessing or post-processing or a new code architecture. The steps typically performed during the qualification process of a new revision or of a new version of the code are depicted in Figure 5. The needed reference data are derived by the following sources.

(1) Analytical experiments, with separate effect tests and component tests, are used for the development and the validation of closure laws.

(2) System tests or integral tests used to validate the general consistency of the revision. Successive revisions of constitutive laws are implemented in successive versions of the code and assessed.

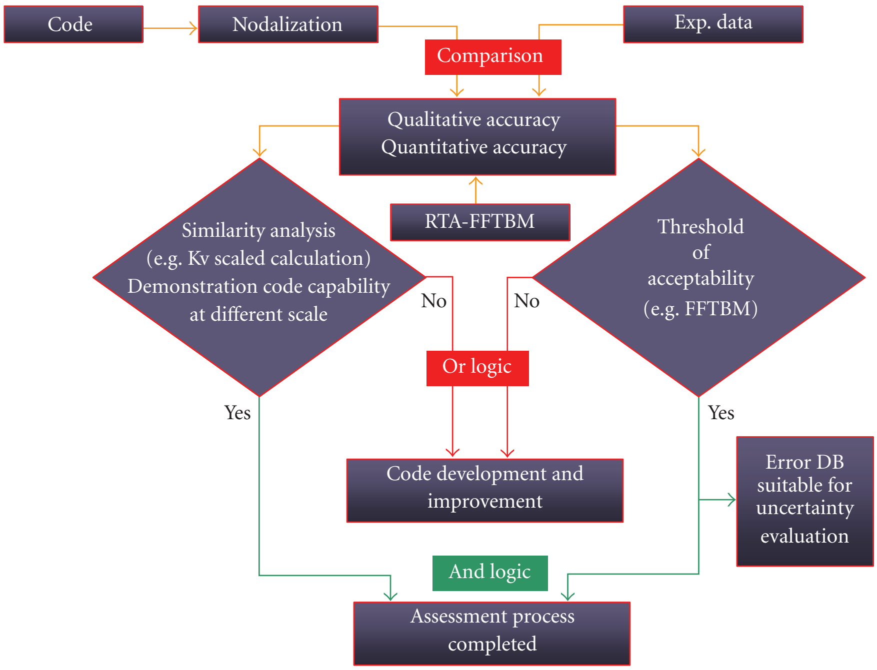

Figure 4: Code independent assessment.

Constitutive relationships are developed and assessed following a general methodology hereafter summarized.

Step A

Analytical experiments, including separate effect tests and component tests, are performed and analyzed. Separate effect tests investigate a physical process such as the interfacial friction, the wall heat transfer. Component tests investigate physical processes which are specific to a reactor component, such as the phase separation in a Tee junction.

Step B

Development of a complete *revision* of constitutive laws from a large analytical experimental data base. Successive revisions are implemented in successive code *versions*.

Step C

Qualification calculations of the analytical tests are used in order to validate each closure relationship.

Step D

Verification calculations of system tests or integral tests are used in order to validate the general consistency of the revision.

Step E

Delivery of the code version and revision is fully assessed (qualified and verified) and documented (description documents and assessment reports).

A new revision of constitutive laws is developed using some general principles.

(1) Data are first compared with existing models; if necessary, original models are developed.

(2) When and where data are missing, simple extrapolations of existing qualified models are used. No mechanistic model is developed without the experimental evidence of its relevance.

(3) In a prequalification phase, some tests of each experiment of the qualification matrix are calculated.

(4) A systematic qualification of the frozen revision is then performed. All tests of the qualification matrix are calculated and qualification reports are written.

Some other additional remarks about the qualification process of the code are as follows.

(1) The qualification program has to cover the whole range of accidental transients in LWR. As examples, the following accidents have to be considered for a PWR: large break loss of coolant accidents (LBLOCA); small break loss of coolant accidents (SBLOCA); steam generator tube ruptures (SGTR); loss of feed water (LOFW); main stream line break (MSLB); loss of residual heat removal (RHR) system.

(2) The code has to be fully portable on all machines, so that a unique code version is released to all the users.

(3) No code options for physical models, or as few as possible, have to be proposed to the user.

(4) The users guidelines should be as precise as possible and take full benefit of the experience gained from the code-development team.

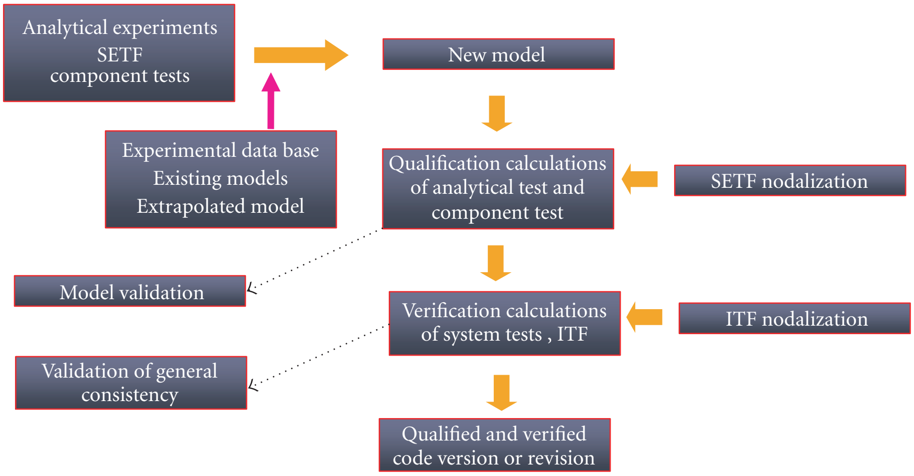

Figure 5: Qualification process of a new revision or a new version of the code.

3.2. *Validation activities for thermal-hydraulic system codes*

The validation against experimental data is essential in the process of system codes development and improvement as it has been discussed in the previous section. The models implemented and used in a code are generally developed based on experimental tests performed in specific facilities. It is possible to distinguish among.

(1) Basic facilities: In these facilities the fundamental phenomena are reproduces; the results are used to improve the equations of the single model or to derive empirically the relation between the relevant parameters; this kind of facilities are designed with goal to reproduce the specific phenomenon to be investigate.

(2) Separate effect facilities: in these facilities some relevant zones of the NPP are reproduced by a suitable scaling law to investigate the local occurrence of a phenomenon; the results of the experiments performed in these facilities are used to create and to validate the (several) models to be included in a code.

(3) Integral tests facilities: these facilities are simulators of reference NPP. All the relevant parts and systems of an NPP are reproduced by a suitable scaling law. The whole plant is reproduced and the global plant response is obtained as results. The results are used to realize and improve the models and to check the code capabilities.

It will be noted that also the data from NPP can be used, if available. However, in an NPP the data obtained are the one recorded by the system of control of the plant while, typically, the facilities are equipped with a large number of sensors and many detailed data are generated making the instrumentation of the facilities more suitable for code validation.

Huge effort was done by the OECD/NEA/CSNI from 1991 to 1997 in the construction of the separate effects test facility code validation matrix (SETF-CCVM, published in 1994) for thermal-hydraulic system codes [3]. Integral test facility (ITF) matrices for validation of realistic thermal-hydraulic system computer codes were also established by CSNI focused mainly on PWRs, and BWRs. The ITF-CCVM [4] validation matrix was issued in 1987 and updated in 1996.

By the validation matrices, the best sets of openly available experimental data for code validation, assessment, and improvement were collected in a systematic way. Quantitative code assessment with respect to the quantification of uncertainties in the modeling of individual phenomena by the codes is also an outcome of the matrix development. In addition, the construction of such matrices is an attempt to record information of the experimental work which has been generated around the world over the last years in the LWR safety thermal-hydraulics field. 187 facilities covering 67 relevant phenomena for LOCA and non-LOCA transient applications of PWRs and BWRs within a large range of useful parameters were identified and about 2094 tests were included in the SETF-CCVM matrix. The majority of these phenomena are also relevant to advanced water-cooled reactors. The major elements of the SETF-CCVM have been already integrated into the validation matrices of the major best-estimate thermal-hydraulic system codes, for example, RELAP5, CATHARE, TRACE, and ATHLET.

A total number of 177 PWR and BWR integral tests have been selected as potential source for thermal-hydraulic code validation in the ITF-CCVM report. Counter-part tests, similar tests and OECD ISP tests were introduced in the report. Counter-part tests and similar tests in differently scaled facilities are considered highly important for code validation and therefore they were included in the tables of ITF selected experiments. Moreover, over the last twenty-nine years, CSNI has promoted 48 ISPs [18]. The main objectives of the ISPs are as follows: to contribute to better understanding of postulated events, to compare and evaluate the capability of codes (mainly best estimate codes), to suggest improvements to the code developers, to improve the ability of code users and to address the so-called scaling effect. ISPs were performed in different fields as in-vessel thermal-hydraulic behavior, fuel behavior under accident conditions, fission product release and transport, core/concrete interactions, hydrogen distribution and mixing, containment thermal-hydraulic behavior. ISP experiments were carefully controlled, documented, and evaluated.

3.3. Addressing the scaling issue

The reason why this section has been included in the paper directly derives from the fact that the scaling analysis is the needed link between the experiments performed in ITF and SETF and their utilization in the code validation process. The flow diagram in Figure 6 emphasizes this relevant role of the scaling analysis (red boxes) in two different parts of the process describing a consistent application (development, qualification, and application) of a thermal-hydraulic system code: firstly during the code assessment process (as the code development and improvement is based on experimental data obtained in test facilities), secondly during the demonstration of the qualification of an NPP nodalization (which is a needed step to perform a reliable NPP calculation).

An NPP is characterized by high power (up to thousands of MW), high pressure (tens of MPa), and large geometry (hundreds of m^3), thus it is well understandable the impossibility to perform experiments preserving all these three quantities. The term scaling is in general understood in a broad sense covering all differences existing between a real full size plant and a corresponding experimental facility. An experimental facility may be characterized by geometrical dimension and shape, arrangements, and availability of components, or by the mode of operation (e.g., nuclear versus electrical heating). All these differences have the potential to distort an experimental observation precluding its direct application for the design or operation of the reference plant. Distortion can be defined as a partial or total suppression of physical phenomena caused by only changing the size (geometric dimension) or the shape (arrangement of components) of the facility [19].

Three main objectives can be associated to the scaling analysis as follows:

(1) the design of a test facility;
(2) the code validation, that is, the demonstration that the code accuracy is scale independent;
(3) the extrapolation of experimental data (obtained into an ITF) to predict the NPP behavior.

For the test facility design, three types of scaling principles can be adopted as follows.

(a) Time-reducing scaling: rigorous reduction of any linear dimension of the test rig would result in a direct proportional reduction in time scaling. This is considered to be of advantage only for cases where body forces due to gravity acceleration are negligible compared to the local pressure differentials.
(b) Time preserving scale: based on a scale reduction of the volume of the loop system combined with a direct proportional scaling of energy sources and sinks (keeping constant the core power to system volume ratio).

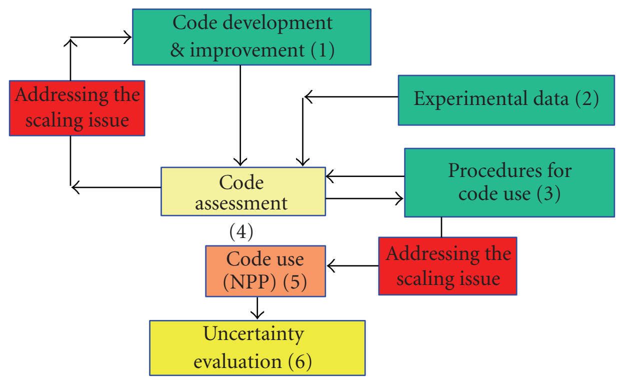

Figure 6: Role of the scaling analysis in the code assessment process.

(c) Idealized time preserving modeling procedures: based on the equivalency of the mathematical representation of the full size plant and of the test rig. It is deduced from a separated treatment of the conservation equations for all involved volume modes and flow paths assuming homogeneous fluid.

Integral test facilities are normally designed to preserve geometrical similarity with the reference reactor system. Generally all main components (e.g., rector pressure vessel, downcomer, rod bundle, loop piping, etc.) and the engineered safety system (HPIS, LPIS, accumulators, auxiliary feed water, etc.) are represented. ITF are used to investigate, by direct simulation, the behavior of an NPP in case of off-normal or accident conditions. The geometrical similarity of the hardware of the loop systems has been abandoned in favor of a preservation of geometric elevations, which are decisive parameters for gravity dominated scenarios (e.g., in case of natural circulation processes). Thus the reduction of the primary system volume is largely achieved by an equivalent reduction in vertical flow cross sections.

Due to the impossibility to perform relevant experiment at full scale (i.e., in an NPP), the use of ITF or SETF is unavoidable. In order to address the scaling issue, different approaches have been proposed and are available from literature. However, a comprehensive solution has not yet been achieved and moreover, it is evident that the attempt to scale up all thermal-hydraulic phenomena that occur during an assigned transient results in a myriad of factors which have counterfeiting values [20]. For instance, let us consider Figure 7 that schematically reproduces a two-phase flow condition (TPFC) in a vessel of a facility when an SBLOCA scenario is postulated. The two-phase critical flow is affected by phenomena like the vapor pull through and the subcooled vapor formation by the sharp edge cavitations, the heat losses, the fluid temperature stratification, and so forth. All these phenomena cannot be scaled up and are characterized by parameters that do appear neither in any balance equations nor in any scalable mechanistic models. This is a typical situation in which a scaling criterion is not applicable. Nevertheless the influence of those phenomena is time-restricted in relation to the entire transient and thus they can be considered as local phenomena.

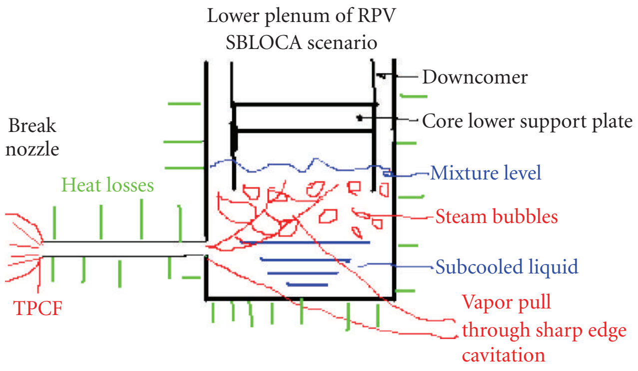

Figure 7: Schematic representation of a two-phase flow condition in a reactor pressure vessel of a facility during an SBLOCA.

As a consequence, the only way to solve the scaling problem is to consider only those phenomena and parameters that have a real impact on the whole problem under investigation. The focusing on a single phenomenon which occurs during a limited time (compared with the entire duration of the problem) should be avoided because it is governed by factors that are not scalable. Therefore a hierarchy in the definition of the scaling factors is necessary and a global strategy is needed [21] to demonstrate that those phenomena are effectively local and cannot affect the overall behavior of the main thermal-hydraulic parameters selected to describe the transient. Based on the flow diagram in Figure 6, the strategy to adopt for solving the scaling problem consists in

(a) developing a system code;
(b) qualifying the code against experimental data;
(c) demonstrating that the code-accuracy (i.e., discrepancy between measured and calculated trends) only depends upon boundary initial conditions (BIC) values (within the assigned variation ranges) and is not affected by the scale of concerned ITF;
(d) applying such code to predict the same relevant phenomena that are expected to find in a same experiment (or transient) performed at different scale;
(e) performing NPP Kv-scaled calculation and explaining the discrepancies (if any) between NPP Kv-scaled calculation and measured trends in ITF considering only BIC values and hardware differences (i.e., distortions).

3.4. Nodalization qualification

Assuming the availability of a qualified code and of a qualified user, it is necessary to define a procedure to qualify the nodalization in order to obtain qualified (i.e., reliable) calculation results. In this section a procedure for the nodalization qualification is discussed.

A major issue in the use of mathematical models is constituted by the model capability to reproduce the plant or facility behavior under steady-state and transient conditions. These aspects constitute two main checks for which acceptability criteria have to be defined and satisfied during the nodalization-qualification process. The first of them is related to the geometrical fidelity of the nodalization of the reference plant; the second one is related to the capability of the code nodalization to reproduce the expected transient scenario.

The checks about the nodalization are necessary to take into account the effect of many different sources of approximations, like the following.

(1) The data of the reference plant available to the user are typically non exhaustive to reproduce a perfect “schematization” of the reference plant.
(2) From the available data, the user derives an approximated nodalization of the plant reducing the level of detail.
(3) The code capability to reproduce the hardware, the plant systems and the actuation logic of the systems reduce further the level of detail of the nodalization.

The reasons for the checks about the capability of the code nodalization to perform the transient analysis deriving from following considerations:

(1) the code options must be adequate;
(2) the nodalization solutions must be adequate;
(3) some systems components can be tested only during transient conditions (e.g., ECCS that are not involved in the normal operation).

A simplified scheme of a procedure that can be adopted for the qualification of the nodalization is depicted in Figure 8 [22]. In the following, it has been assumed that the code has fulfilled the validation and qualification process and a “frozen” version of the code has been made available to the final user. This means that the code user does not have the possibility to modify or change the physical and numerical models of the code (only the options described in the user manual are available to the user). With reference to Figure 8, the qualification procedure of the nodalization is described step by step.

Step “a”

This step is related to the information available by the user manual and by the guidelines for the use of the code. This type of information takes into account the specific limits and assumptions of the code (specific of the code adopted for the analysis) and some guidelines deriving from the best practices for realizing the nodalization. From a generic point of view, the following aspects should be carefully adopted:

(1) homogeneous nodalizations;
(2) strict observation of the user guidelines;
(3) standard use of the code options.

Step “b”

User experience and developers recommendations are useful to set up particular procedure to be applied for a better nodalization. These special procedures are related to the specific code adopted for the analysis. An example is constituted by the "slice nodalization" technique adopted with the RELAP5 code to improve the capability of the code to reproduce transients involving natural circulation phenomena.

Step "c"

The realization of the nodalization depends on several aspects: available data, user capability and experience, code capability. The nodalization must reproduce all the relevant parts of the reference plant; this includes geometrical and materials fidelity and reproduction of the systems and related logics. From a generic point of view, the following recommendations can be done.

(1) Data must be qualified or in other words, data has to derive from
  (a) qualified data facility (if the analysis is performed for a facility);
  (b) qualified test design;
  (c) qualified test data.

(2) The data base for the realization of the nodalization should be derived from official document and traceability of each reference should be maintained. However three different types of data can be identified as follows:
  (a) qualified data, from official sources;
  (b) data deriving from nonofficial sources; these types of data can be derived from similar plant data, or other qualified nodalization for the same type of plant; the use of these data can introduces potential errors and the effect on the calculation results must be carefully evaluated;
  (c) data assumed by the user; these data constitute some assumptions of the user (on the base of the experience or by similitude with other similar plants). The use of this type of data should be avoided. Any special assumptions adopted by the user or special solutions in the nodalization must be recorded and documented.

Step "d"

The "steady-state" qualification level includes different checks: one is related to the evaluation of the geometrical data and of numerical values implemented in the nodalization; the other one is related to the capability of the nodalization to reproduce the steady-state qualified conditions. The first check should be performed by a user different from the user has developed the nodalization. In the second check a "steady-state" calculation is performed. This activity depends on the different code peculiarities. As an example, for RELAP5, the steady-state calculation is constituted by a "null-transient" calculation (i.e., the "transient" option is selected and no variation of relevant parameters occurs during the calculation).

Step "e"

The relevant geometrical values and the relevant thermal-hydraulic parameters of the steady-state conditions are identified. The selected geometrical values and the selected relevant parameters are derived, respectively, from the input deck of the nodalization and from the steady-state calculation for performing the comparison with the hardware values and the experimental parameters.

Step "f"

This is the step where the adopted acceptability criteria are applied to evaluate the comparison between hardware and implemented geometrical values in the nodalization (e.g., volumes, heat transfer area, etc.) and between the experimental and calculated steady-state parameters (e.g., pressures, temperatures, mass flow rates, etc.). Some comments can be added as follows.

(1) The experimental data are typically available with error bands which must be considered in the comparison with the calculated values and parameters.
(2) The steadiness of the steady-state calculation must be checked.

Step "g"

If one or more than one of the checks in the step "f" are not fulfilled, a review of the nodalization (step "c") must be performed. This process can request more detailed data, improvement in the development of the nodalization, different user choices. The path "g" must be repeated till all acceptability criteria are satisfied. A list of the geometrical values and of the thermal-hydraulic parameters to be checked is given in Table 1 together with acceptable errors.

Step "h"

This step constitutes the "On Transient" level qualification. This activity is necessary to demonstrate the capability of the code nodalization to reproduce the relevant thermal-hydraulic phenomena expected during the transient. This step also permits to verify the correctness of some systems that are in operation only during transient events. Criteria, both qualitative and quantitative, are established to express the acceptability of the transient calculation. Two different aspects can be identified as follows.

(1) The code input deck concerns with the nodalization of an ITF. In this case the code calculation is used for the code assessment. Checks include the code options selected by the user, the solutions adopted for the development of the ITF nodalization, the logic of some systems (e.g., ECCS). Typically many experimental results are available, thus a similar test can be adopted for performing the "On Transient" level qualification.

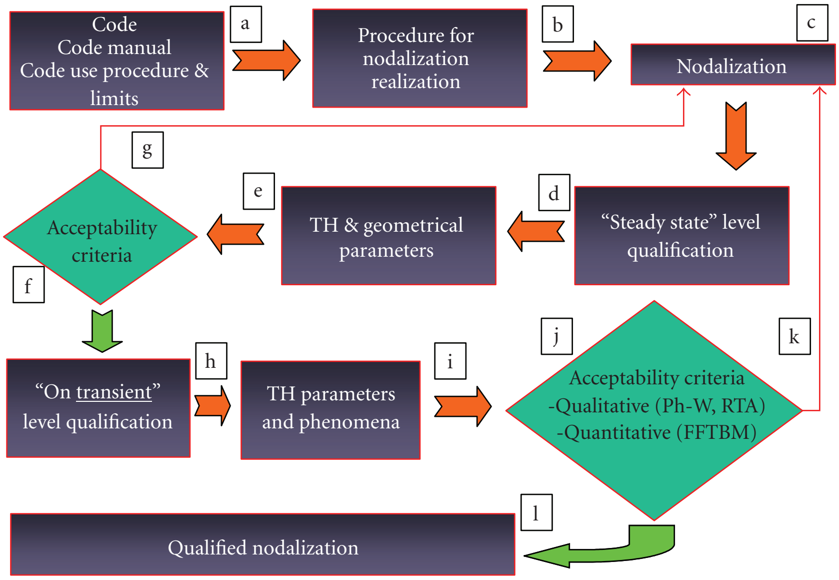

FIGURE 8: Flow sheet of nodalization qualification procedure.

(2) The objective of the code calculation is constituted by the analysis of a transient in an NPP. In this case, it is necessary to check the nodalization capability to reproduce the expected thermal-hydraulic phenomena occurring during the transient, the selected code options, the adopted solutions for the development of the NPP nodalization, and the logic of the systems not involved in the steady-state calculation. Typically no data exist for the transients performed in the NPP. For this reason, data from experiments carried out in ITF can be used for performing the so-called "Kv-scaled" calculation. The Kv-scaled calculation consists in using the developed NPP nodalization for predicting an experimental transient (whose kind is similar to the one under investigation in the NPP) performed in an ITF. The NPP nodalization is prepared for the Kv-scaled calculation by properly scaling the BICs characterizing the selected transient in the ITF. In other words, power, mass flow rates and ECCS capacity are scaled adopting as scaling factor the ratio between the volume of the facility and the volume of the NPP. The capability of the nodalization to reproduce the same transient evolution and the thermal-hydraulic relevant phenomena is the needed request for satisfying the "On Transient" qualification level.

Step "i"

In this step the relevant thermal-hydraulic phenomena and parameters are selected and a comparison between the calculated and experimental data is performed. The selection of the phenomena derives from the following sources:

(1) experimental data analysis (engineering judgment is request);
(2) CSNI phenomena identification;
(3) use of Relevant Thermal-hydraulic Aspects (RTA, engineering judgment is request).

Step "j"

This is the step where checks are performed to evaluate the acceptability of the calculation both from qualitative and from quantitative point of view. For the qualitative evaluation the following aspects are involved:

(1) Visual observation. This means that a visual comparison is performed between experimental and calculated relevant parameters time trends;
(2) Sequence of the resulting events. This means that the list of the calculated significant events together with their timing of occurrence is compared with the experimental events;
(3) Use of the CSNI phenomena. The relevant phenomena suitable for the code assessment and their relevance in the selected facility and in the selected test are identified. A judgment can be express taking into account the characteristics of the facility, the test peculiarities and the code results;
(4) Use of the RTAs. RTAs are typically identified inside the phenomenological windows (i.e., time windows where a unique relevant phenomenon is occurring) and are characterized by special parameters. These parameters can be time values, single values, integral values, gradient values and nondimensional values. An example of a table containing RTAs is given in Table 2.

Quantitative checks are carried out by using the Fast Fourier Transform Based Method (FFTBM). This special tool performs the comparison between experimental and calculated time trends in the frequency domain for a list of selected parameters and calculates, for each of them, a numerical value by which the accuracy is quantitatively evaluated (no engi-

Table 1: Parameters and acceptable errors for the nodalization qualification at "steady-state" level.

	Quantity	Acceptable error (°)
1	Primary circuit volume	1%
2	Secondary circuit volume	2%
3	Nonactive structure heat transfer area (overall)	10%
4	Active structure heat transfer area (overall)	0.1%
5	Non-active structure heat transfer volume (overall)	14%
6	Active structure heat transfer volume (overall)	0.2%
7	Volume versus height curve (i.e., "local" primary and secondary circuit volume)	10%
8	Component relative elevation	0.01 m
9	Axial and radial power distribution (°°)	1%
10	Flow area of components like valves, pumps orifices	1%
11	Generic flow area	10%
(*)		
12	Primary circuit power balance	2%
13	Secondary circuit power balance	2%
14	Absolute pressure (PRZ, SG, ACC)	0.1%
15	Fluid temperature	0.5% (**)
16	Rod surface temperature	10 K
17	Pump velocity	1%
18	Heat losses	10%
19	Local pressure drops	10% (^)
20	Mass inventory in primary circuit	2% (^^)
21	Mass inventory in secondary circuit	5% (^^)
22	Flow rates (primary and secondary circuit)	2%
23	Bypass mass flow rates	10%
24	Pressurizer level (collapsed)	0.05 m
25	Secondary side or downcomer level	0.1 m (^^)

° The % error is defined as the ratio (reference or measured value—calculated value). The "dimensional error" is the numerator of the above expression.
* With reference to each of the quantities below, following a one-hundred-second "null-transient" calculation, the solution must be stable with an inherent drift <1%/100 second.
** And consistent with power error.
^ Of the difference between the maximum and minimum pressure in the loop.
^^ And consistent with other errors.

neering judgment is involved in this process). The FFTBM makes also possible to obtain a numerical judgment of the overall results of the calculation. Criteria based on the values attained by FFTBM had been selected for accepting the transient calculation. A description of the FFTBM can be found in [23].

Step "k"

This path is actuated if any of the checks (qualitative and quantitative) is not fulfilled. The nodalization is improved by adopting different noding solutions, changing code options or increasing the level of detail using, if available, more precise data. Every time the nodalization is modified a new qualification process will be performed through the loop "c-d-e-f-h-i-j-c."

Step "l"

This is the last step of the procedure. The obtained nodalization is used for the selected transient and the selected facility or plant. Any subsequent modification of the nodalization (e.g., necessary to better reproduce the experimental results) requires a new qualification process both at "steady-state" and "on transient" level.

4. DEVELOPMENT AND USE OF COUPLED COMPUTER CODES

Complex computer codes are used for the analysis of the performance of NPPs. They include many types of codes that can be grouped in different categories [24] like reactor physics codes; fuel behavior codes; thermal-hydraulic codes, including system codes, subchannel codes, porous media codes and computational fluid dynamic (CFD) codes; containment analysis codes; atmospheric dispersion and dose codes and structural codes.

Historically, these codes have been developed independently, but have been mainly used in combination with system thermal-hydraulic codes. By increasing the capacity of computation technology, safety experts thought of coupling these codes in order to reduce uncertainties or errors associated with the transfer of interface data and to improve the accuracy of calculation. The coupling of primary sys-

Table 2

		UNIT	EXP	UNIPI91BN1OLPSI	CEAc2m4_lcea	JudgmentUNIPI/CEA
RTA: pressurizer emptying						
TSE	Emptying time*	s	131	46	—	R/-
	Scram time	s	41	38	41	R/E
RTA: steam generators secondary side behaviour						
TSE	Main feed water off, turbine bypass	s	59	55	42	E/R
SVP	Difference between PS and SG 1 SS pressure at 100 s	MPa	0.42	0.33	0.37	R/R
SVP	SG 1 mass	Kg/(s)				
	at the end of subcooled blowdown		774/(82)	781/(75)	761/(82)	E/E
	when PS pressure equals SG 1 SS pressure		869/(618)	938/(408)	847/(463)	R/R
	when ACC starts		804/(2955)	802/(3019)	788/(3075)	E/R
	when LPIS starts		938/(5176)	1126/(6529)	956/(5474)	R/R
SYP	SG 1 pressure	MPa				
	at the end of subcooled blowdown		7.15	7.10	7.05	E/E
	when PS pressure equals SS pressure		6.95	7.04	7.03	R/R
	when ACC starts		4.11	3.95	4.00	R/E
	when LPIS starts		0.88	0.83	0.83	E/E
RTA: subcooled blowdown						
TSE	Upper plenum in sat conditions	s	83	100	110	R/R
IPA	Break flow up to 100 s	kg	152	161	162	R/R
RTA: first dryout occurrence						
TSE	Time of dryout	s	2237	2299	2444	E/R
	Range of dryout occurrence at various core levels	s	2237÷2471	2299÷2518	2444÷2625	R/R

tem thermal-hydraulics with neutronics is a typical example of code coupling; other cases include coupling of primary system thermal-hydraulics with structural mechanics, fission product chemistry, computational fluid dynamics, nuclear fuel behavior and containment behavior. Problems that need to be addressed in the development and use of coupled codes include ensuring adequate computer capacity and efficient coupling procedures, validation of coupled codes and evaluation of uncertainties, and consequently the applicability of coupled codes for safety analyses.

The major purposes of the development of coupled code are to be capable of representing the results of interactions between different physical phenomena in more detail. Since the calculation method of each code is not changed, reduction of computational time or necessary computer memory volume is not expected. Nevertheless, many additive benefits are expected as follows.

(1) Since the interface data are easily, automatically and frequently exchanged between codes, the results of calculation would be obtained faster than the combination of individual codes and also be more reliable.

(2) Since the development works are limited to the interface part, the cost and time for development can be minimized.

(3) Since the interface data between each code would be adjusted to meet the specifications (e.g., noding of the system or time increment of calculation) of each code at the development stage, additional assumptions or data averaging and reductions are not required when performing the calculation.

(4) Those that have the knowledge of the existing codes are not necessary to study the coupled code from the beginning, because the existing knowledge is applicable to the coupled code.

It is expected that those benefits can contribute to the improvement of activities carried out by both licensing authorities and industries. Expectations for licensing authorities can mainly be derived from the features of coupled codes such as more accurate calculation than the combination of individual codes. These are summarized as follows:

(i) improvement of the understanding of the phenomena of interest for safety;

(ii) better assessment/demonstration of the conservatisms (versus historical approaches such as the use of point kinetics or evaluation models);

(iii) extension of the capabilities of the codes for safety analysis and training/simulators;

(iv) better assessment of uncertainties associated with the use of best estimate couplet codes.

Many benefits are expected with the use of coupled codes for industries. These are as follows.

(i) Faster turnaround of calculation allows the users to perform more precise analysis and more sensitivity or case studies. This would contribute in more detail to understand the features of the plant, systems or components.
(ii) More accurate calculation would contribute to remove unnecessary uncertainties and to identify margins available to use for the plant.
(iii) Uncertainties due to user effects would be minimized because the existing knowledge of individual codes is applicable to the coupled codes.

The request to use qualified tools in licensing calculations constitutes one of the main problems to be addressed in the development of coupled computer codes and it is caused by the limited availability of data, which can be obtained from operating plants. To reduce the effort for the qualification of the coupled codes, code developers are requested to use only validated revisions of codes. In addition, the code developers are requested to

(i) design the coupling so that auditing is easy and feasible;
(ii) provide guidelines to minimize user effects;
(iii) allow provisions for reasonable conservatisms;
(iv) structure the code so that coupling is easy and feasible;
(v) standardize the coupling procedures;
(vi) integrate as much as possible the existing approved calculation methodologies.

5. CONCLUSIONS

A noticeable progress in the capabilities of system codes has been observed in the past decades. From the design and safety engineering point of view, thermal-hydraulic system codes are considered to have reached an acceptable level of maturity. Most of the problems and questions that come up a couple of decades ago have been solved or an answer has been proposed. In other words, there is more need to synthesize the work done in the international ground than to identify new problems. For instance, if corresponding measured and calculated trends are given, possible research should be focused on answering whether the discrepancy is acceptable and less on minimizing the discrepancy itself (e.g., through an improved model). It is evident that all the progress has been made in the recent past is a consequence of experimental researches. After 30 years of validation through basic, separate and integral effect tests facilities and after code improvements, system codes are able to predict main phenomena of PWR & BWR transients with reasonable accuracy. Nowadays the attention should be focused more on developing procedures for a consistent application of a thermal-hydraulic system code. This need has been highlighted in the paper and implies the drawing up of specific criteria through which the code-user, the nodalization and finally the calculated transient results can be qualified.

The full exploitation of "advanced" best-estimate system codes (e.g., TRAC, RELAP, ATHLET, CATHARE), which are strictly based on two-fluid representation of two-phase flow and a "best-estimate" description (in contrast with the evaluation models which used many conservative assumptions) of complex flow and heat transfer conditions, implies mainly their acceptability by the licensing authorities. In fact, notwithstanding the important achievements and progresses made in the recent years, the predictions of advanced best-estimate computer codes are not exact but remain uncertain because of the following.

(i) The assessment process depends upon data almost always measured in small scaled facilities and not in the full power reactors.
(ii) The models and the solution methods in the codes are approximate: in some cases, fundamental laws of the physics are not considered.

Consequently, the results of the best estimate code calculations may not be applicable to give "exact" information on the behavior of an NPP during postulated accident scenarios. Therefore, best-estimate analysis must be supplemented by proper uncertainty evaluations in order to be meaningful and conditions for their application should be made clear for accepting the available uncertainty methods in the licensing process.

In conclusion, the present status, of system codes development, assessment, and related uncertainty evaluation, is adequate as far as the largest majority of design and safety problems of current water-cooled reactors are concerned. Anyway, new scientific goals must be achieved. To this aim, projects and programmes based on the development of system codes with multidimensional and multifluid capability and with "open" interfaces for an easy coupling with other codes in areas like neutronics (for implementing presently available 3D codes), CFD, structural mechanics (e.g., for pressurized thermal-shock studies), and containment constitute the new frontier of the scientific and engineering community in this field. However, taking into account that the development of such codes with measurable increased improvements in their capabilities may need several decades, it is an evident consequence that the existing system thermal-hydraulic codes are going to be used for one or two decades in their present configuration.

ABBREVIATIONS

1D, 3D:	One-dimensional, three-dimensional
BE:	Best estimate
BEMUSE:	Best-estimate methods-uncertainty and sensitivity evaluation
BIC:	Boundary initial conditions
BWR:	Boiling water reactor
CCVM:	CSNI code validation matrix
CFD:	Computational fluid dynamic
CSAU:	Code scaling applicability and uncertainty
CSNI:	Committee on the safety of nuclear installations
ECCS:	Emergency core cooling systems
FFTBM:	Fast fourier transform based method
HEM:	Homogeneous equilibrium model
HPIS:	High pressure injection system
ISP:	International standard problem

ITF: Integral test facility
LBLOCA: Large break loss of coolant accidents
LOCA: Loss of coolant accident
LOFW: Loss of feed water
LPIS: Low pressure injection system
LWR: Light water reactor
MSLB: Main steam line break
NPP: Nuclear power plants
OECD: Organization for cooperation and development
PSA: Probabilistic safety analysis
PWR: Pressurized water reactor
RHR: Residual heat removal
RTA: Relevant thermal-hydraulic aspect
SBLOCA: Small break loss of coolant accidents
SETF: Separate effect test facility
SGTR: Steam generator tube ruptures
SQA: Software quality assurance
TPFC: Two-phase flow condition
UMS: Uncertainty method study

REFERENCES

[1] "Acceptance criteria for emergency core cooling systems (ECCS) in light water nuclear reactors (10CFR 50.46)," Appendix K to Part 50 "ECCS Evaluation Models", Federal Register, vol. 43, no. 235 (43 FR 57157), December, 1978.

[2] S. Belsito, F. D'Auria, and G. M. Galassi, "Application of a statistical model to the evaluation of counterpart test database," *Kerntechnik*, vol. 59, no. 3, 1994.

[3] N. Aksan, N. D'Auria, H. Glaeser, R. Pochard, C. Richards, and A. Sjoberg, "A separate effects test matrix for thermal-hydraulic code validation: phenomena characterization and selection of facilities and tests," OECD/GD (94) 82, vols. I and II, 1993.

[4] N. Aksan, D. Bessette, I. Brittain, et al., "Code validation matrix of thermo-hydraulic codes for LWR LOCA and transients," CSNI Report 132, Paris, France, March 1987.

[5] M. J. Lewis, , R. Pochard, F. D'Auria, et al., "Thermohydraulics of emergency core cooling in light water reactors-a state of the art report," OECD/CSNI 161, October 1989.

[6] USNRC, "Compendium of ECCS research for realistic LOCA analysis," NUREG-1230, December 1988.

[7] B. E. Boyack, I. Catton, R. B. Duffey, et al., "Quantifying reactor safety margins—part 1: an overview of the code scaling, applicability, and uncertainty evaluation methodology," *Nuclear Engineering and Design*, vol. 119, no. 1, pp. 1–15, 1990.

[8] N. Aksan, F. D'Auria, H. Glaeser, R. Pochard, C. Richards, and A. Sjoberg, "Overview of the CSNI separate effects test validation matrix," in *Proceedings of the 7th International Topical Meeting on Reactor Thermal Hydraulics (NURETH '95)*, New York, NY, USA, September 1995.

[9] S. N. Aksan, F. D'Auria, and H. Stdtke, "User effects on the thermal-hydraulic transient system code calculations," *Nuclear Engineering and Design*, vol. 145, no. 1-2, pp. 159–174, 1993.

[10] M. Bonuccelli, F. D'Auria, N. Debrecin, and G. M. Galassi, "A methodology for the qualification of thermalhydraulic codes nodalizations," in *Proceedings of the 5th International Topical Meeting on Reactor Thermal Hydraulics (NURETH '93)*, Grenoble, France, October 1993.

[11] F. D'Auria, M. Leonardi, and R. Pochard, "Methodology for the evaluation of thermalhydraulic codes accuracy," in *Proceedings of International Conference on New Trends in Nuclear System Thermalhydraulics*, Pisa, Italy, May-June 1994.

[12] F. D'Auria and W. Giannotti, "Development of code with capability of internal assessment of uncertainty," *Nuclear Technology*, vol. 131, no. 1, pp. 159–196, 2000.

[13] A. Petruzzi, F. D'Auria, W. Giannotti, and K. Ivanov, "Methodology of internal assessment of uncertainty and extension to neutron kinetics/thermal-hydraulics coupled codes," *Nuclear Science and Engineering*, vol. 149, no. 2, pp. 211–236, 2005.

[14] T. Wickett, et al., "Report of the uncertainty method study for advanced best estimate thermal-hydraulic code applications," vols. I and II, in OECD/CSNI Report NEA/CSNI R (97) 35, Paris, France, 1998.

[15] A. Petruzzi, et al., "BEMUSE Programme. Phase 2 report (re-analysis of the ISP-13 exercise, post test analysis of the LOFT L2-5 experiment)," in OECD/CSNI Report NEA/CSNI/R(2006)2, pp. 1–625, 2006.

[16] A. De Crecy, et al., "BEMUSE Programme. Phase 3 report (uncertainty and sensitivity analysis of the LOFT L2-5 experiment)," in OECD/CSNI Report NEA/CSNI/R(2007)4, 2007.

[17] F. D'Auria and G. M. Galassi, "Code validation and uncertainties in system thermalhydraulics," *Progress in Nuclear Energy*, vol. 33, no. 1-2, pp. 175–216, 1998.

[18] N. Aksan, "International standard problems and small break loss-of-coolant accident (SBLOCA)," in *Proceedings of THICKET: Seminar on Transfer of Competence, Knowledge and Experience Gained Through CSNI Activities in the Field of Thermalhydraulics, CSNI OECD/NEA, INSTN and IRSN*, Saclay, France, June 2004.

[19] F. D'Auria, K. Fischer, B. Mavko, and A. Sartmandjiev, "Validation of accident and safety analysis methodology," Internal Technical Report, International Atomic Energy Agency, Vienna, Austria, June 2001.

[20] N. Zuber, G. E. Wilson, M. Ishii, et al., "An integrated structure and scaling methodology for severe accident technical issue resolution: development of methodology," *Nuclear Engineering and Design*, vol. 186, no. 1-2, pp. 1–21, 1998.

[21] F. D'Auria, G. M. Galassi, and P. Gatta, "Scaling in nuclear system thermal hydraulics: a way to utilise the available database," in *Proceedings of the 32nd National Heat Transfer Conference (ASME '97)*, vol. 350, pp. 35–43, Baltimore, Md, USA, 1997.

[22] A. Petruzzi, F. D'Auria, and W. Giannotti, "Description of the procedure to qualify the nodalization and to analyze the code results," DIMNP NT 557(05), May, 2005.

[23] A. Petruzzi and F. D'Auria, "Accuracy quantification: description of the fast fourier transform based method (FFTBM)," DIMNP NT 556(05), May, 2005.

[24] IAEA-TECDOC-1539 (2007), "Use and development of couplet computer codes for the analysis of accidents at nuclear power plants," in *International Atomic Energy Agency, Technical Meeting*, Vienna, Austria, November 2003.