STATE OF THE ART IN USING BEST ESTIMATE CALCULATION TOOLS IN NUCLEAR TECHNOLOGY

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System thermal-hydraulic codes have been used in the past decades in the areas of design, operation, licensing and safety of Nuclear Power Plants (NPPs). The development and validation of these codes have reached a high degree of maturity, through the consideration of huge experiments and advanced numerical models. Nowadays, the analyses are based upon realistic approaches rather than the conservative evaluation models. However the applications of these computational tools require preliminary qualification issues. Although huge amounts of financial and human resources have been invested for the development and improvement of codes, the calculation results are still affected by errors. In the sophisticated nuclear technology, design and safety of NPP, these errors must be quantified. An overview of the state of the art of the current thermal-hydraulic system code is developed and the need of uncertainty analysis in code calculations is emphasized. Several sources of uncertainty have been classified and commented, and typical applications of such methods are shown.

KEYWORDS : NPP Accident Analysis, Best Estimate in System Thermal-hydraulics, Uncertainty, Quality of Predictions, User Effect, Scaling, Coupled Code

1. INTRODUCTION

Computer codes are widely used for NPP safety analysis within a wide set of purposes including licensing issues, safety improvement programs of existing NPPs, better utilization of nuclear fuel, higher operational flexibility, for justification of lifetime extensions, development of new emergency operating procedures, analysis of operational events, and development of accident management programs. A safety key parameter of the evaluation and assessment of NPPs is closely related to the code ability of determining the time-space thermal-hydraulic conditions throughout the reactor coolant system and especially in the core region. At the beginning, the code development took place between the sixties and seventies where sets of conservative models are used. These latter were limited due mainly to the restricted computer memory, CPU time, and performances [1]. However, in the light of the sustained development in computer technology and computational methods, the potential of the codes has been enlarged substantially. Nowadays, it becomes possible to switch to new generation of computational tools by coupling advanced computer codes. This will allow getting better realistic simulations of complex phenomena and transients that could occur in NPP [2]. The coupled code technique includes mainly thermal-hydraulic system and reactor kinetics codes, as well as specific codes for the containment thermal-hydraulics, structural mechanics codes, and more sophisticated Computational Fluid Dynamics (CFD) codes.

However, notwithstanding the complexity of these codes and the level of the present scientific knowledge, a computer code cannot be expected to accurately model phenomena that are not yet fully understood by the scientific community. In general, the results of code predictions, specifically when compared with experimental data reveal often some discrepancies. These discrepancies could be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, imperfect knowledge of boundary and initial conditions. Therefore, it is necessary to investigate the uncertainty of the results and the sensitivity effect of the most effective parameters. Following some pioneering work, promoted by the US Nuclear Regulatory Commission and leading to, the proposal of the CSAU (Code Scaling, Applicability and Uncertainty) [3], different uncertainty methodologies have been developed by different research organisations in order to evaluate the reliability of any thermalhydraulic code calculation, taking into account the possible sources of errors.

The purpose of the present paper is to characterize the present situation as far as the code assessment and uncertainty predictions are concerned. This is achieved through a re-evaluation of the activities carried out at the University of PISA (UPISA), including the participation of the authors in a number of international projects. On this basis, requirements and future needs in the field of thermal-hydraulic system codes are obtained.

2. THE FRAMEWORK OF THERMALHYDRAULIC SYSTEM CODES

2.1 Historical Perspective

Since early '60s, until today, the Thermal-Hydraulic System Codes (THSC) have undergone deep changes and substantial improvements. This has imposed among other things, a continuous assessment process leading to the latest released code versions. In this period, a large number of facilities, that have been classified as Integral Test Facilities (ITF) and Separate Effect Test Facilities (SETF) have been designed and put into operation all over the world and specifically in Italy. This has led to the availability of a huge amount of experimental data that was used for qualifying the codes and for identifying and characterizing new phenomena, thus requiring additional code improvements. This process used large amounts of resources till the beginning of '90s. Items in Table 1 constitute recent milestones in the nuclear reactor thermal-hydraulic technology and give an idea of the main achievements.

Historically, three technical areas exist so far, dealing respectively with the "primary loop" (within Design Basis Accident (DBA) and beyond DBA before loss of geometric integrity of the core), the "containment" (characterization of thermalhydraulic scenarios mostly in the same conditions as above) and the "severe accident" (in-vessel and ex-vessel system behaviour, corium behaviour, hydrogen

Table 1. List of the Main Activities in the Uncertainty Analysis

YEAR	ACTIVITY
1980-1982	Scaling analysis for the design of the PIPER-ONE BWR simulator
1982-	Proposal for design criteria for PIPER-ONE
1985	Analysis of SBLOCAs in PWR on the basis of 'similar' tests
1985	Proposal for criteria for accuracy quantification
1987	**Publication of OECD/CSNI ITF-CCVM**
1988	Proposal of criteria for planning 'Counterpart' tests (CT)
1989	**Issue of US NRC Compendium on ECCS Research**
1989	**Issue of OECD/CSNI on TECC**
1989	'Use' of CT data related to BWRs
1989-1992	Papers dealing with the basis of the UMAE uncertainty methodology
1990	**Publication of CSAU**
1990	Studies on user effect, bringing to a CSNI publication in 1992
1990	Proposal for the FFTBM for accuracy quantification
1990	Analysis of Natural Circulation in PWR on the basis of 'similar' tests
1991	Proposal for a methodology for independent assessment of codes
1992	Analysis of LOFW in PWR on the basis of similar tests
1992	Proposal for a procedure for nodalisation qualification
1993	Simplified flow-sheet of UMAE and differences with respect to CSAU
1993	Analysis of SBLOCA in PWR on the basis of performed CT
1993	Application of UMAE to a SBLOCA in Krsko PWR
1993	**Publication of OECD/CSNI SETF-CCVM**
1994	**Completion of the 2D-3D Research Program and planning of TRAM**
1995	Issue of UMAE-ET (to account for 'unrecoverable' code errors)
1995	Comparison between features of uncertainty methodologies
1996	Issue of UMAE-SETF (to exploit SETF data)
1996	**Publication of OECD/CSNI on Lesson Learned from SBLOCA ISP**
1996	Proposal for a procedure for code user training, see also (6)
1997	Application of UMAE utilising Relap5/mod2 and Cathare 2v1.3 codes
1997	Application of UMAE to Angra-1 PWR
1997	Proposal for the CIAU (idea at the basis of the method)
1998	**Publication of OECD/CSNI UMS report**
1997-1999	Execution of different Kv scaled calculations
1999	Demonstration of feasibility of CIAU and preliminary results
2000	**Publication of IAEA Guidelines for Accident Analysis–Draft**
2001	Bifurcation analysis and CIAU matrix enlargement Application of CIAU to Angra-2 and Kozloduy-3 NPP LBLOCA
2002	Development of uncertainty for 3-D neutronics/ thermalhydraulics coupled codes
2005	**OECD-BEMUSE Program (Best-Estimate Methods Uncertainty and Sensitivity Evaluation)**

production, diffusion, etc.). Limited links among these three areas existed till the advent of the advanced reactor where, among other things, the consideration of a tight coupling between primary system and containment was considered necessary to predict the overall system performance. In the above context, three finalizations for code use at the UPISA, can be outlined:

- Characterization of transient scenarios in nuclear power plants;
- Code assessment including the participation in International Programs;
- Application of codes developed for the design or licensing of nuclear power plants.

In addition, an important part of the research in the area has been dedicated to the development and the qualification of the simulations errors of a NPP related transient scenarios [3], [5], [6], [7].

2.2 Framework of the THSC

The established method to evaluate complex conditions throughout the interconnected systems of a given NPP is carried out by the so-called Best Estimate (BE) THSC, e.g. RELAP5, TRAC, ATHLET, and CATHARE. Due to the numerical approximations and of the empirical nature of included models in the thermal-hydraulic system codes extensive activities related to validation of the code models have been pursued during the years. The validation has partly been done 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:

a. "Fundamental" experiments [8];
b. Separate Effects Test Facilities (SETF) [9];
c. Integral Test Facilities (ITF), including most of the International Standard Problems [10];
d. Real plant data [11], [12], [13]

An additional level for code assessment can be identified including the so-called numerical benchmarks, also covering the demonstration of suitability of the adopted numerical solution scheme. This can be considered as belonging to the developmental assessment [14], and the OECD/NEA Benchmarks as the PWR Main Steam Line Break (MSLB) in TMI-1 [15].

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

- The codes have reached an acceptable degree of maturity although the reliable application is still limited to the validation domain;
- The codes availability is increasingly growing especially in the Countries belonging to the former Soviet Union, the Eastern Countries, Korea, China, etc.;
- 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;
- Code validation criteria and detailed qualification programs exist, although not fully optimized or internationally agreed;
- Methodologies to evaluate the 'uncertainty' (i.e. the error) in the prediction of nuclear power plants related scenarios by system codes have been proposed and are being tested;
- Problems like user effect (i.e. influence of code users in the predictions), nodalization qualification, quantification of code accuracy (i.e. ranking of the error in the comparison between measured and calculated trend) have been dealt with and experience is currently available;
- Activities have been recently completed that are coordinated by the OECD Committee on the Safety of Nuclear Installations (CSNI), the OECD/NEA, or by the International Atomic Energy Agency (IAEA). These activities concern mainly the assessment of computer code models and predictions [16], [17], [18], and [19].

2.3 Qualification Process

The basic idea of the THSC is to perform a numerical description of a plant (or facility) and the considered fluid stream-tubes and solid structures (slabs) that store heat or contain heat sources, within specified boundaries and assumptions. Each portion of the plant of interest is divided (in a number of ways) into discrete components (nodes) called control volumes. The degree of details of the nodalization depends mainly on the code features and its numerical restrictions.

Four fundamental pre-conditions shall be fulfilled for the correct application of a complex thermal-hydraulics system code to the prediction of transient scenarios expected in NPP:

- The code should be frozen.
- The code should be properly qualified through wide, preferably international, assessment programs.
- The developer of the nodalization should be a qualified code user for the selected code [4] and [20].
- The nodalization of the plant once developed, should be properly qualified [21] and [22].

Each portion of the plant that is of interest for the analyses is divided into discrete components (nodes). The model is developed by the process of dividing the real plant component volumes into a set of control volumes that are essentially stream-tubes having inlet and outlet flow path connections. It is clear that subdivision of such a complex system as LWR plant can be done in a number of ways. The simplest subdivision of a plant model would be into a set of control volumes or nodes that are equally sized, but for a successful solution in the case of the analysis, a number of factors must be satisfied: numerical stability, run time, and spatial convergence. In addition, engineering judgment is normally used to a wide extent to develop an input deck. The importance of establishing a procedure for the nodalization set-up and qualification is a consequence. Such a procedure can be split into the following steps [2]:

a) Gathering of a verified set of NPP data
b) Set-up of the plant nodalization (input deck for the nominal steady state conditions),
c) Qualification of the nodalization.

2.4 Development and Qualification Procedure for Nodalization

Plant nodalization should be developed according to the predefined list of qualitative criteria outlined in Table 2 [23]. For instance geometrical fidelity with the real modeled system should be kept, all fluid flow paths shall be modeled, logics for normal and off-normal operating conditions should be included, the sliced (or sandwich type) nodalization-concept should be followed, as far as possible. Therefore, to achieve a reliable nodalization the following items should be fulfilled:

1) The nodalization should have a geometrical fidelity with the involved plant.

Table 2. Acceptability Criteria for Thermal-hydraulic Nodalization Qualification at 'Steady-state' Level

	QUANTITY	ACCEPTABLE ERROR (°)
1	Primary circuit volume	1%
2	Secondary circuit volume	2%
3	Non-active structures heat transfer area (overall)	10%
4	Active structures heat transfer area (overall)	0.1%
5	Non-active structures heat transfer volume (overall)	14%
6	Active structures heat transfer volume (overall)	0.2%
7	Volume vs. 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 and orifices	1%
11	Generic flow areas	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 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 value - calculated value| / |reference value|. The dimensional error is the numerator in the above expression.
(*) With reference to each of the quantities below, following a 100 s 'transient - steady state' calculation, the solution must be stable with an inherent drift < 1% / 100 s.
(**) And consistent with power error.
(^) Of the difference between maximum and minimum pressure in the loop.
(^^) And consistent with other errors.

2) The nodalization should reproduce the nominal measured steady-state condition of that plant. In the steady state level, the operational parameters of the simulated system should fulfill the acceptability criteria as outlined in [23]. A sketch of the overall process of nodalization qualification at the steady state and transient levels, and improvement can be found in Fig.1.

3) The nodalization should get a satisfactory behavior in time-dependent conditions of any test performed, or operational transients in the nuclear power plant. In fact, the demonstration of the nodalization quality at the steady state level does not ensure that the prediction of a transient scenario is 'phenomenologically' correct or even that the nodalization (input deck) is free from errors.

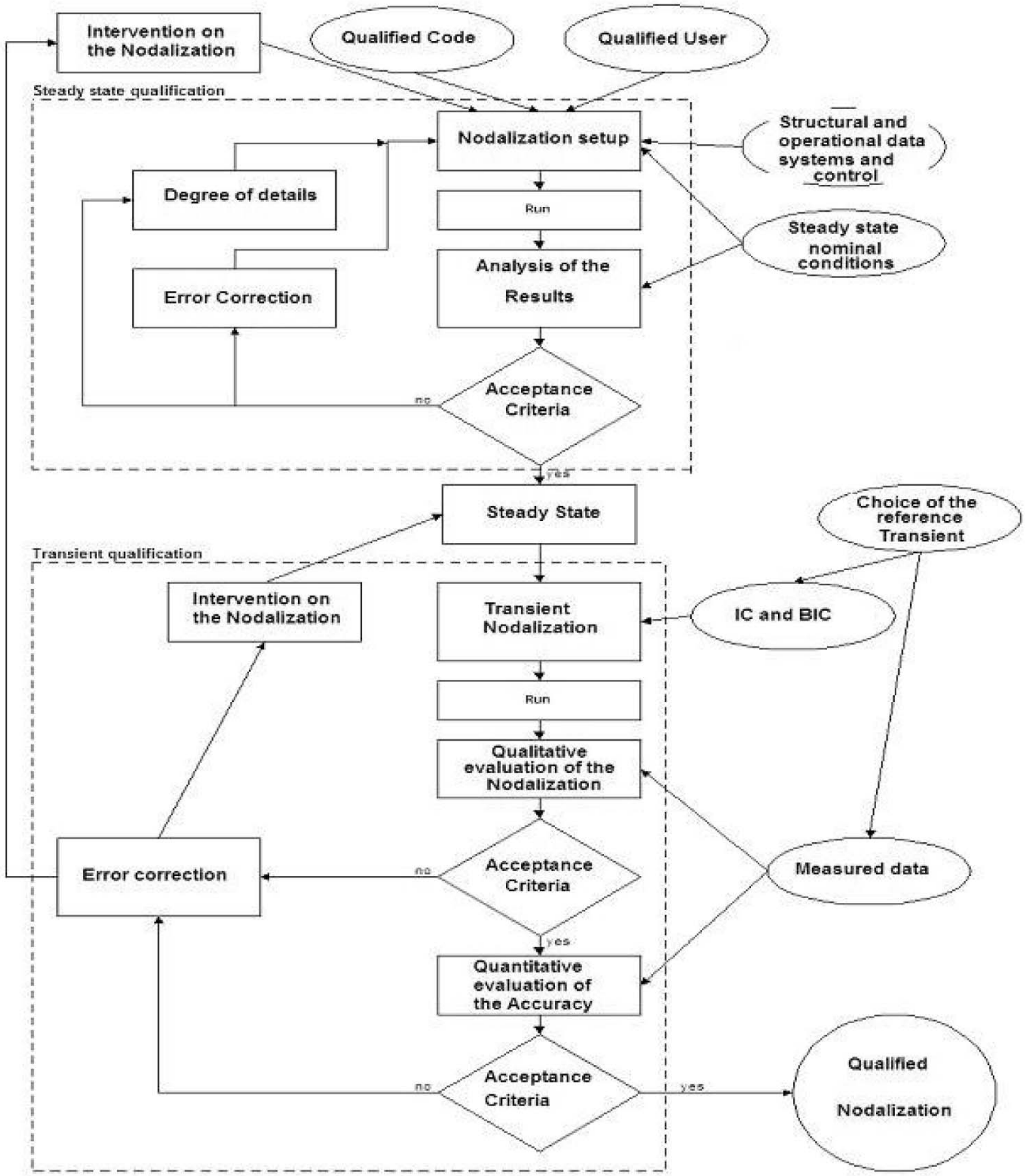

Fig. 1. Thermal-hydraulic Nodalization Qualification Process

Errors can be part of an input deck that has been qualified at the 'steady-state' level. The on-transient nodalization qualification process is demonstrated through the capability to correctly predict (with acceptable discrepancies) relevant phenomena and transient scenarios of the facility being simulated. On the other hand, quality demonstration for the output of the 'Kv-scaled' calculation is obtained from the qualitative and quantitative accuracy evaluation adopting suitable analytical tools [5], and [24].

2.5 The Fast Fourier Transform Based Method (FFTBM) Method

Several approaches have been proposed to quantify the accuracy of a given code calculation [25], [26], and [27]. Even though these methods were able to give some information about the accuracy, they were not considered satisfactory because they involved some empiricism and were lacking of a precise mathematical meaning. Besides, engineering subjective judgment at various levels is deeply inside in proposed methods. Generally, the starting point of each method is an error function, by means of which the accuracy is evaluated. Some requirements were fixed which an objective error function should satisfy:

1) At any time of the transient this function should remember the previous history;
2) Engineering judgment should be avoided or reduced;
3) The mathematical formulation should be simple;
4) The function should be non-dimensional;
5) It should be independent upon the transient duration;
6) Compensating errors should be taken into account (or pointed out);
7) Its values should be normalized.

The simplest formulation about the accuracy of a given code calculation, with reference to the experimental measured trend, is obtained by the difference function:

$$\Delta F(t) = F_{calc}(t) - F_{\exp}(t) \tag{1}$$

The FFTBM characterizes each calculation through two values:

- A dimensionless average amplitude, AA:

$$AA = \frac{\sum_{n=0}^{2^m} \left|\tilde{\Delta} F(f_n)\right|}{\sum_{n=0}^{2^m} \tilde{F}_{\exp}(f_n)} \tag{2}$$

- A weighted frequency, WF:

$$WF = \frac{\sum_{n=0}^{2^m} \left|\tilde{\Delta} F(f_n)\right| \cdot f_n}{\sum_{n=0}^{2^m} \tilde{\Delta} F(f_n)} \tag{3}$$

The most significant information is given by AA, which represents the relative magnitude of the discrepancy deriving from the comparison between the addressed calculation and the corresponding experimental trend (AA=1 means a calculation affected by a 100% of error). The WF factor characterizes the kind of error, because its value emphasizes whether the error has more relevance at low or high frequencies, and depending on transient, high frequency errors can be more acceptable than low frequency ones (in other words, analyzing thermal-hydraulic transients, better accuracy is generally represented by low AA values at high WF values.

Trying to give an overall picture of the accuracy of a given calculation, average indexes of performance are obtained by defining:

$$(AA)_{tot} = \sum_{i=1}^{N_{\text{var}}} (AA) \cdot (w_f)_i \tag{4}$$

With

$$\sum_{i}^{N_{\text{var}}} (w_f)_i = 1 \tag{5}$$

Where N_{var} is the number of analyzed parameters and $(w_f)i$ are weighting factors that take into account the different importance of each parameter from the viewpoint of safety analyses.

Following the quantitative evaluation of accuracy, the Quantitative Assessment (QA) can be managed by means of the application of the FFT method. Obviously, the most suitable factor for the definition of an acceptability criterion is the average amplitude AA. With reference to the accuracy of a given calculation, we can define the following acceptability criterion:

$$(AA)_{tot} < K \tag{A.8}$$

Where K is an acceptability factor that is valid for the whole transient. As lower is the AAtot value, as better is the accuracy of the analyzed calculation. With reference to experience gathered from previous applications of this methodology, K = 0.4 has been chosen as reference threshold value identifying acceptable accuracy of a code calculation.

3. KEY FEATURES OF THERMALHYDRAULIC SYSTEM CODES

Practical purposes or objectives of the THSC are identified in the following [3]:

· Licensing process. A BE code should erode the conservatism of previous generation Evaluation Models. In other words, making reference to the classical hot rod surface temperature trend versus time, the situation should be as follows: Best-Estimate Plus Uncertainty (BEPU) should be lower than the Evaluation model result and stay well below the allowed limit. Positive consequences for the industry could be the relaxation of current requirements, e.g. one Low Pressure Injection Pump may reveal sufficient instead of two or design requirements (head and flow) for two pumps may be relaxed, with advantages in maintenance or initial system cost, respectively;
  - Possibility of upgrading power of the plant;
  - Use of a unique code for design, maintenance and licensing. The team using the conservative code to fulfill requirements of the licensing authority is not needed.

· Safety analyses. Safety is clearly connected with licensing; nevertheless, safety analyses can be conducted outside the licensing process. An example is constituted by the safety evaluations of existing reactors of Soviet origin [26]. A versatile, qualified, and publicly available tool must be used to this aim; a BE code is the only applicable tool.

· Design of new plants. The design of the majority of existing plant, at least in relation to the main hardware features, was completed in the 60's without the help of existing codes. Nevertheless BE codes have been used for design confirmation. Envisaged use in the area is:

· Design optimization: e.g. number of U-tubes of SG, position and number of recirculation pumps, volume of pressurizer and of the vessel downcomer, etc.;

· Design of 'passive' reactors: effectiveness of emergency systems is more difficult to demonstrate than in current generation reactors; the use of THSC codes appears mandatory.

· Optimization of Emergency Operating Procedures (EOP). Findings from the operation of experimental facilities, from the results of BE codes, from Accident Management (AM), and Probabilistic Safety Assessment (PSA) related studies, opened new possibilities in this area. Achievement of THSC tools is being mandatory for demonstrating the suitability and the applicability or any new EOP.

· Operator training and simulators qualification. The training, of operators, the part done through plant simulators, needs realistic accident scenarios and NPP feedback to operator interventions. This can only be achieved through BE codes that must be used to benchmark the simpler codes at the basis of the simulator in all conditions of interest. Existing post processors, including advanced graphical interfaces allow the direct use of BE codes for operator training.

3.1 Safety Margins, Sensitivity and Uncertainty

One of the objectives of safety analysis is to provide a robust demonstration that all safety requirements are met, i.e. that sufficient margins exist between real values of important parameters and their threshold values at which damage of the barriers against release of radioactivity would occur. The concept of safety margins is introduced in Fig.2. As can be seen from the figure there are two possibilities to define safety margins: either in absolute terms, in relation to expected damage of safety barriers or in relation to acceptance criterion, typically set up by the regulatory body. Fig.2 also illustrates the difference between results of conservative and best estimate analysis. While in conservative approach, the results are expressed in terms of a set of calculated conservative values, in best estimate approach the results are expressed in terms of uncertainty ranges for each of the calculated parameters. On the other hand, the Safety Guide (SG) on safety assessment recommends performing both sensitivity and uncertainty analysis. It is important to underline that sensitivity analysis must not be misinterpreted as code uncertainties. Sensitivity analysis means evaluation of the effect of arbitrary variation in input or modeling parameters on code results, while uncertainty analysis means a statistical combination of code uncertainties, representation uncertainties and plant data uncertainties. These two analyses may coincide only under very special conditions.

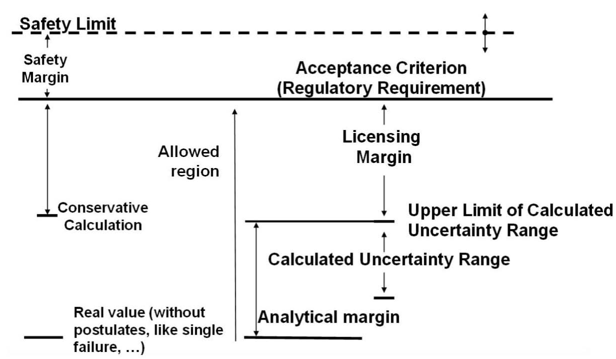

Fig. 2. Illustration of Safety Margins

3.2 Conservative Approach Versus Uncertainty Evaluation

The word 'uncertainty' and the need for uncertainty evaluation are connected with the use of BE codes instead of 'conservative' tools or assumptions in the code application. The application of THSC codes implies the choice of the BE approach. Table 3 summarizes various options for combining computer codes and input data for safety analysis. Two different categories of data are

Table 3. Various Options for Combination of a Computer Code and Input Data

Option	Computer cod	Availability of systems	Initial and boundary conditions
1	Conservative	Conservative assumptions	Conservative input data
2	Best estimate	Conservative assumptions	Conservative input data
3	Best estimate	Conservative assumptions	Realistic input data with uncertainties
4	Best estimate	PSA based assumptions	Realistic input data with uncertainties

distinguished: assumptions on availability of plant systems (normal operation systems, control systems, safety systems) and on all other initial and boundary conditions.

Totally conservative approach (option 1) was introduced to cover uncertainties at the level of knowledge in the 1970s. However, the results obtained by this approach may be misleading (unrealistic behavior predicted, order of events changed) and level of conservatism is unknown. Therefore, use of this approach is now unwarranted. Options 2 and 3 are considered as acceptable and suggested according to the existing IAEA Safety Standards. Option 2 is more typically used at present for safety analysis. It is reasonably well established and its use seems to be straightforward: in many cases just one calculation is sufficient to demonstrate safety. Comparison to the uncertainty estimates developed within the framework of international code validation as well as in the studies on representation and plant data uncertainties and various sensitivity studies help to establish confidence in robustness in the predicted NPP behavior.

Conservative modeling approaches can be still used to avoid the cost of developing a realistic model. However, this approach provides only some estimate of the uncertainties, many calculations are often needed to support conservative selection of input data and still intentional conservative may not lead to conservative results. An example is assumption of high power during small-break loss of coolant accident (SBLOCA), which over-predicts swell level in the core and this leads to better core cooling, opposite to conservative requirement. Different sets of conservative assumptions are typically required for each of the acceptance criteria, and even different assumptions may be needed for different parts of a transient.

Best estimate analysis provides a good picture of the existing safety margins. Unfortunately, pure best estimate approach is not always possible or desirable because of the difficulty of quantifying code uncertainties with sufficiently narrow range for every phenomenon and for each accident sequence. It is important to see, that even though this represents one more step towards pure best estimate analysis, there is still a significant conservative component in this approach; namely due to usual conservative assumptions on availability of NPP systems (e.g. non-availability of control systems in accident situations, single failure criterion, combination with loss of power supply in some cases, etc.).

4. UNCERTAINTY METHODS

Uncertainty analyses include the estimation of uncertainties in individual modeling or of the overall code, uncertainties in representation, and uncertainties in plant data for the analysis of an individual event. Scaling studies to quantify the influence of scaling variations between experiments and the actual plant environment are included in this definition. In some references, code scaling and uncertainty analysis are identified separately.

4.1 Origin of Uncertainties

Thermal-hydraulic system code calculations are affected by unavoidable errors arising from several causes, including the unavoidable approximations in the constitutive equations, from the limited capabilities of numerical solution methods, from uncertainties in the knowledge of boundary and initial conditions, from errors in setting up the nodalization.

These can be characterized by hundreds of parameters that are typically part of the input deck for a system code calculation suitable for predicting a transient scenario in a Nuclear Power Plant. This happens notwithstanding the high code performance level and the systematic qualification processes, nowadays in progress or completed. It is necessary to remind that the user choices strongly affect the code results, through the so called "user effect" [4].

Following some pioneering work, promoted by the US Nuclear Regulatory Commission, the proposal of the UMAE (Code Scaling, Applicability and Uncertainty), [27] to [29], different uncertainty methodologies have been developed by different research organisations, [30] in order to evaluate the reliability of any thermal-hydraulic code calculation, taking into account the possible sources of error. The system thermal-hydraulic codes, whose results are subject to uncertainty evaluations, are far from being perfect; their solutions are approximate. The characterization and/or the origin of the approximations constitutes the goal of uncertainty studies. Reference is made to the use or the features of two system codes (RELAP5 and CATHARE) that are in use at the UPISA, but can be extended to other codes. A THSC deals with the solution of balance equations based upon first principles of physics supplemented by empirical correlations; these are numerically solved utilizing boundary and initial conditions supplied by code user. The code itself generally consists of more than hundred of Fortran statements and the input decks may reach hundreds of 'cards', so errors can easily be part of such packages of electronic statements. An indicative list of approximations is given hereafter, making reference to steam-liquid mixtures.

A) Balance or conservation equations are approximate.
B) Presence of different fields of the same phase. Liquid in a two-phase mixture may be present in the form of droplets or film (e.g. two fields) that are characterized by different velocities, temperatures. Only one velocity, temperature, etc., is calculated by the current codes.
C) Geometry averaging at cross-section scale. Different velocity profiles happen in the reality depending upon local values of thermodynamic quantities and upon history. Averaged values are provided by codes that are valid in a limited number of situations.
D) Geometry averaging at volume scale. One velocity is associated to a volume or hydraulic mesh in the main fluid direction, i.e. assumed axis for motion.
E) Large and small vortex or eddy simulation. In single and two-phase flow, unavoidably vortices and eddies appear with different modalities depending upon the geometric and thermodynamic conditions. These create energy and momentum dissipation not directly accounted for by codes.
F) The second principle of thermodynamics is not necessarily fulfilled by the solved equations that basically deal with the simulation of irreversible phenomena. During the simulation of an irreversible process inside an isolated system, entropy may not be predicted to increase.
G) Unavoidable numerical truncation and round off errors.
H) Correlations implementation and range of validity may not be fully specified.
I) 'Steady State' and 'Fully Developed' (SS & FD) flow approximations.
J) State and material properties are approximate.
K) Code user effect that may interact at different levels with code results, as pointed out in [4].
L) Computer/compiler effects. A code installed in any computer machine should produce the same results provided a unique input deck is adopted. This is not the case due to a number of reasons connected with the precision of the machine and with the compiler design, [1], [31].
M) Nodalization effects which tends to homogenize the complex systems.
N) Imperfect knowledge of Boundary or Initial Conditions (BIC).
O) Code/model deficiencies cannot be excluded. Such deficiencies could appear only in special transient situations. However, they constitute an additional specific source of uncertainty.

Hereafter, the above-mentioned detailed sources of uncertainty have been associated with the three broad uncertainty sources early indicated, supplemented by two additional ones [32]:

1) **Code Uncertainty** : A THSC is a computational tool that typically includes three different sets of balance equations (or of equations derived from fundamental principles), closure or constitutive equations, material and state properties, special process or component models and a numerical solution method. The sources of uncertainty connected with the code are those identified as A) to I) and O) in the above list. Namely, the following association between uncertainty sources and code parts applies:
   - Balance equations: uncertainty sources A) to F).
   - Closure and constitutive equations: uncertainty sources H) and I).
   - Material properties: uncertainty source J).
   - Special process and component models: Uncertainty sources H), I) and O).
   - Numerical methods: uncertainty source G).
2) **Representation Uncertainty** : The sources of uncertainty connected with the nodalization is identified as M) in the above list, but the J) source can also have a role.
3) **Scaling** : Almost all code models are evaluated from tests at reduced scale. This requires some logical assessment of effects of scale. These occur at the level of fundamental phenomena, where numerous dimensionless groups can be derived. The sources of uncertainty connected with the scaling are those applicable to the balance equations, e.g. identified as A) to I) in the above list. More precisely uncertainty sources associated to the scaling are A) to E), H) and I). The uncertainty associated with scaling may be attributed to insufficiently 'uncertainty-driven' code assessment process.
4) **Plant Uncertainty** : The source of uncertainty connected with the plant is identified in N).
5) **User Effec t** : Complex systems codes such as ATHLET, CATHARE, RELAP5, and TRAC have many degrees of freedom that encourage misapplication and errors by users. Two competent users will not approach the analysis of a problem in the same way and consequently, will likely take different paths to obtain a problem solution. The sources of uncertainty connected with the code-user are those identified as K) and J). The code user has part of the responsibility associated with the source of uncertainty L).

4.2 Approaches Used for the Evaluation of Uncertainty

An uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as by the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). The approaches pursued for uncertainty evaluation can be distinguished into two main categories, i.e. propagation of code-input uncertainties and of code-output errors, respectively. From the mathematical point of view, methods fully based upon statistics can also be distinguished from fully deterministic methods where the expertise of "uncertainty-methodology-user" is needed at different steps to achieve meaningful results. The attention is focused hereafter toward the methodologies that were submitted to a deep review process in the frame of the UMS (Uncertainty Method study) promoted by the CSNI. These have been proposed by AEAW (Winfrith, UK), ENUSA (Madrid, Spain), IPSN (Cadarache, France), GRS (Munich, Germany), and University of Pisa (Pisa, Italy) and are identified in the following as AEAW, ENUSA, IPSN, GRS and UMAE (Uncertainty Methodology based on Accuracy Extrapolation), respectively. A common aspect for all the methodologies is constituted by the use of experimental data in the process; however, modalities in the use and type of needed data are different. Basically, three approaches have been distinguished:

4.2.1 Purely Deterministic

Any use of statistics is avoided; results of various steps of the methodology are checked and evaluated by the methodology user. The most influencing parameters for an assigned transient are selected together with their ranges of variations. The process must end up with a limited number of code runs. Input parameters are modified that can be accessible or not to code user (e.g. coefficients of correlations embedded into the code can be modified) [33].

4.2.2 Purely Statistics:

The methodology is characterized by the user specified range of variation of each parameter and the number of performed code runs necessary to get uncertainty bounds is a function of the level of confidence in the results. This category is based upon the input propagation error (see Fig.3) that includes mainly the code model approximations.

4.2.3 Based upon the Propagation of Code Output Errors:

Inaccuracies of calculations are characterised by comparing measured and calculated time trends of relevant variables. Inaccuracies of calculations are propagated from the facilities to the reference system: extrapolation of accuracy to get uncertainty. The basic assumption is that relevant experimental data are available and include almost

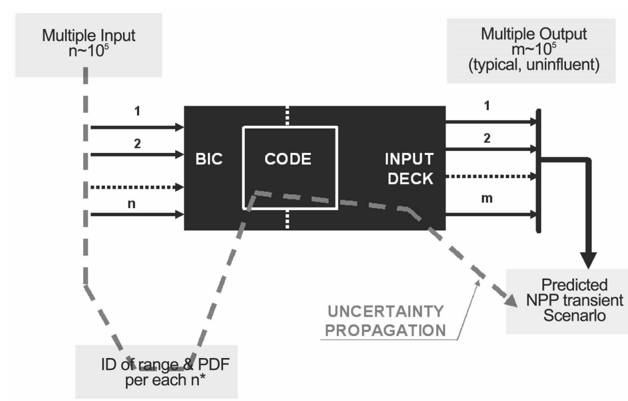

Fig. 3. Uncertainty Method Based Upon Propagation of Input Uncertainties

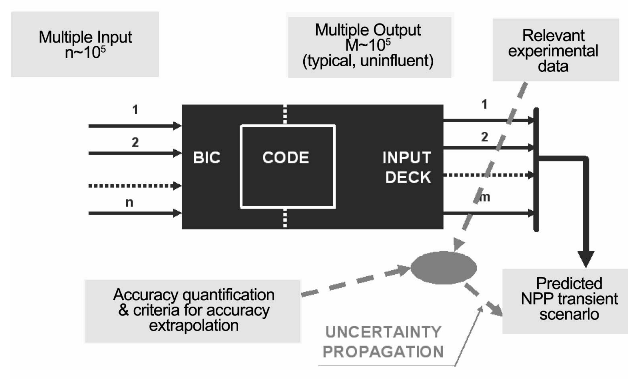

Fig. 4. Uncertainty Method Based Upon Propagation of Output Uncertainties

all the uncertainty sources expected in the reference transient; other uncertainty sources must considered separately through proper biases. The main characteristics of the methods based on the propagation of output uncertainties (see Fig.4) is error extrapolation issued from former simulations of relevant experimental data.

4.3 The Uncertainty Methods Study

The Uncertainty Methods Study (UMS) Group, following a mandate from CSNI, has compared five methods (AEAW, GRS, UMAE, IPSN, and ENUSA) for calculating the uncertainty in the predictions of advanced best estimate thermal-hydraulic codes. The objectives of the international UMS were [3], [34]:

1. To gain insights into differences between features of the methods by:
   - Comparing the different methods, step by step, when applied to the same problem;

Table 4. - Summary of the Compared Methods

	PROPAGATION OF CODE-INPUT UNCERTAINTIES		PROPAGATION OF CODE-OUTPUT ERRORS	
DETERMINISTIC METHODS	AEAW (UK)	Phenomena uncertainties selected, quantified by ranges and combined.		
STATISTICS METHODS	CSAU (US)	Uncertainty of safety-related output single-valued parameters (e.g. PCT). A response surface approach has been followed.	UMAE (University of Pisa) (CIAU)	Accuracy in calculating similar integral tests is extrapolated to plant.
	GRS (Germany)	Phenomena uncertainties quantified by ranges and subjective probability distribution functions (SPDFs) and combined.		
	ENUSA (Spain)	Phenomena uncertainties quantified by ranges and SPDFs and combined.		
	IPSN (France)	Phenomena uncertainties quantified by ranges and SPDFs and combined.		

- Comparing the uncertainties predicted for specified output quantities of interest;
- Comparing the uncertainties predicted with measured values;

2. To inform those who will take decisions on conducting uncertainty analyses, for example in the light of licensing requirements.

In Table 4 a list of the methods, hereafter discussed, is given with respect to the different approaches pursued and with a brief description.

4.3.1 AEAW Method

The uncertainty statements or input uncertainty ranges should be in the form of "reasonable uncertainty ranges". Such a range is defined as "the smallest range of values (of a given quantity) that includes all values for which there is reasonable certainty that they are consistent with all available evidence". The method is planned to select input uncertain and ranges that are consistent with the above statement. Once a qualified code is made available from developers, the code and input uncertainty ranges are used to predict independent data; these must be representative of the processes expected to occur in the plant transient. If the predicted uncertainty ranges bound the selected independent experimental data, the code and the uncertainty analysis can be used for plant calculations. If not, then further development is needed for the code and/or the input uncertainty data, and the data from the old independent database move into the development database.

4.3.2 UMAE Method

The basic idea of UMAE is the use of the accuracy from the comparison between measured and calculated trends of relevant experiments and calculations, respectively. The experiments must come from relevant facilities and the calculation results from qualified codes and nodalizations. This avoids the need to select input uncertainties; also, resulting uncertainty range are coming from the process and do not need subjective evaluations. The development of suitable nodalizations and qualification at the 'steady state' and the 'on-transient' levels are needed. The process of nodalization qualification is fully independent from the process aiming at the derivation of the extrapolated accuracy (different data bases are used). The fulfillment of various conditions (quality of data base, of NPP nodalization, of code performance) allows the finalization of the process that, vice versa, can be interrupted at different stages. In the first situation, accuracy, coming from several comparisons between measured and calculated trends can be "extrapolated" and becomes uncertainty. This is superimposed to the unique best-estimate code run performed by a qualified NPP nodalization.

4.3.3 GRS and IPSN Methods

These two methods are basically the same the only difference being the type of input parameters that are selected in the process. The methods have the capability to consider the effect of uncertainty of input parameters like computer code models, initial and boundary conditions, other application specific input data and solution algorithms on the calculation results. They are based on well-established concepts and tools from probability calculus and statistics. The GRS method is applicable to any computational tool, without modifications of the codes [33]. The analyses are fully based on the statistical variation of model input parameters and the error margins could be obtained for any output parameter of the code.

4.3.4 ENUSA Method

ENUSA method is based on the UMAE. The PIRT (Phenomena Identification and Ranking Table) process is used to select a reasonable number of input uncertain parameters, i.e. the AEAW method. The ranges of variations are fixed, again utilizing the same or a similar approach as AEAW but, in addition, Subjective Probability Distribution Functions (SPDF) are identified by ENUSA method (in a similar way as GRS and IPSN methods) as opposite to the AEAW method where only the ranges of output variables are considered. In order to minimize the number of calculation runs, the process of combination of input uncertainties is basically the same as adopted by GRS and IPSN. This makes the difference between the ENUSA method and the UMAE and may justify considering this method as statistically based.

The common features to the five considered uncertainty are the following:

· Each method has the capability to calculate error ranges as a function of time, i.e. continuous error bands, that bound best estimate code calculation results;
· Each method consists of a limited number of main steps and assumption that appear evident when the method is applied, see also ref. [34];
· Each method requires resources of the order of man-years to be used the first time by a competent (in thermal -hydraulics), technician who is unaware either of the method or of the field of application;
· Some features of each method are directly connected with the adopted code. This is valid to a different extent in the various cases: examples are criteria for developing nodalizations, or selection of input parameters for uncertainty that may not exist in each code;
· Each method requires the selection of a code and of a transient (reference scenario, and reference NPP);
· Each method, as already mentioned, makes experimental data to a different extent;
· Each method needs a qualified code;
· Each method aims at providing information useful to decision makers.

The flow diagrams and the adopted nomenclature for identifying assumptions, are not uniform or consistent among the methods: as an example, the PIRT is used by UMAE, and then by ENUSA, to screen the phenomena; phenomena screening is also necessary in the AEAW method; but the PIRT is not used or even not mentioned. This together with the second starred item above, had to be overcome in the comparison among the methods, as summarised in ref. [34]. Nevertheless, an additional evaluation of the selected uncertainty methods is performed as outlined in Table 5 and 6.

Table 5. Main Characteristics of Uncertainty Methods

	GENERAL CHARACTERISTICS	AEAW	UMAE	GRS	UMAE
1	Restriction on the number of input uncertain parameters	yes	yes	no	n.a.
2	Deriving input uncertainty ranges	yes	yes	yes	no
3	Assigning subjective probability distribution	no	yes	yes	no
4	Use of statistics	no	yes	yes	yes[(a)]
5	Use of response surface technique	no	yes	no	no
6	Necessity of specific data for scaling	no	no[(b)]	no	yes
7	Quantification of code calculation accuracy	no	no	no	yes
8	Use of expert groups	yes	yes	yes	no
9	Use of biases on output	no	yes	no	yes

[(a)] To a limited extent.
[(b)] At a qualitative level, during code validation.

Table 6. Comparison Among Relevant Features of Uncertainty Methods

	GENERAL CHARACTERISTICS	AEAW	UMAE	GRS	UMAE
1	Determination of uncertain parameters and of input uncertainty ranges	expert	expert	expert	(1)
2	Selection of uncertain parameter values within the determined range for code calculations	expert	expert	random selection	expert
3	Support of identification and raking of main parameter and modeling uncertainties	no	yes	no	no
4	Account for state of knowledge of uncertain parameters (distribution of input uncertainties)	no	yes	yes	no
5	Probabilistic uncertainty statement	no	yes	yes	yes
6	Statistical rigor	n.a.	no	yes	no
7	Knowledge of code specifics may reduce resources necessary to the analysis	yes	yes	no	no
8	Number of code runs independent from number of input and output parameters	no	no	yes	no
9	Typical number of code runs	LOBI: 22	LB: 8 SB: 34	59 OMEGA: 100	n.a.[2]
10	Number of uncertain input parameters	LOBI: 7	LB: 7(+5) SB: 8	OMEGA: 60	n.a.[2]
11	Quantitative information about influence of a limited number of code runs	no	no	yes	no
12	Continuous-valued output parameters	yes	no	yes	yes
13	Sensitivity measures of input parameters on output parameters	no	no	yes	no

(1) The differences between experimental and used input data: these constitute one of the sources for uncertainty.

(2) It depends on the stage of the analysis. Fist application to the analysis of the Small Break LOCA counterpart test in PWR required roughly 20 code runs; the analysis of a similar NPP scenario would require roughly a few additional code runs.

<u>In relation to Table 5:</u>

The increase in the number of input uncertain parameters, item 1), causes substantial increases in the needed computational resources for applying AEAW and UMAE: for this reason this number must be minimized. This is not the case for the GRS/IPSN and ENUSA methods. In the case of UMAE, no uncertain input parameters must be specified: whatever is the selected list of parameters (provided their number is sufficiently large, [34]), the uncertain results will not change.

All methodologies require, to different extents, that scaling analysis be made, item 6). This is done at qualitative level during code validation and may require specific analyses during the application of the methodology. In the case of UMAE the demonstration that a given physical phenomenon occurs in differently scaled facilities is necessary: this can be achieved through the use of the FFTBM [27]. Owing to the above, the UMAE could not be applied if experiments reproducing the target test scenario are not available.

Experts are needed by all methodologies, e.g. for the optimal use of the code, for ranking the importance of phenomena or for selecting relevant experimental data. However, expert judgements are not needed by users of UMAE, or if used, their influence is controlled by fixed targets of accuracy. Biases on output values should be avoided or reduced to the minimum extent to prevent the adding of subjective judgements on the output uncertainty, item 9). UMAE considers the biases on output value a cost efficient way to reduce the code runs. In the case of UMAE, calculation of biases may come from the lack of consideration of phenomena expected in the reference transient and not present in the database.

<u>In relation to Table 6:</u>

Prioritization of input parameters is supported by a ranking or brainstorming processes in the case of AEAW and UMAE, item 3). The state of knowledge of input uncertain parameters is expressed, where applicable, by subjective probability distribution, item 4). Expert judgement is needed in the case of GRS and UMAE to account for the state of knowledge. In the case of UMAE, the knowledge of parameter uncertainty is not considered; however, significant error may cause failure of the process. Statistical rigorous algorithms are part of GRS, IPSN and ENUSA methodologies, item 6). This is not the case of UMAE; however the influence of statistics related algorithms upon the results is much reduced in the UMAE compared with the other methods. The statistics based methods, GRS, IPSN and ENUSA allow the possibility to evaluate the influence of a reduced number of calculations, item 11), and of the most important input uncertain parameters on the output uncertainty, item 12). This is not possible in the case of UMAE. A similar situation occurs in relation to the characterization of the relevance of input uncertain parameters, as far as output uncertainty is concerned, item 13).

4.4 The BEMUSE (Best-Estimate Methods and Uncertainty Evaluation) Program

The BEMUSE (Best Estimate Methods – Uncertainty and Sensitivity Evaluation) program, promoted by the working group on accident management and analysis (GAMA) and endorsed by the committee on the safety of nuclear installations (CSNI) [35], represents an important step on the road to the reliable application of high-quality best-estimate and uncertainty evaluation methods. The activity consists in 6 phases subdivided in two steps:

- Step 1: best-estimate and uncertainty evaluation of the LOFT L2-5 test (Phases I, II and II);
- Step 2: best-estimate sensitivity studies and uncertainty evaluation for a NPP- Large Break Loss Of Coolant Accident (LBLOCA) (Phases IV, V and VI).

The operational objective of the activity is the quality demonstration of the system code calculations in performing LBLOCA analysis through the fulfillment of a comprehensive set of common criteria established in correspondence of different steps of the code assessment process.

In particular criteria and threshold values for selected parameters have been adopted for:

a) The developing of the nodalization;
b) The evaluation of the steady state results;
c) The qualitative and quantitative comparison between measured and calculated time trends.

The technological importance of the activity can be derived from the following:

a) LOFT is the only ITF with a nuclear core where safety experiments have been performed;
b) The ISP-13 was completed more than 20 years ago and open issues remained from the analysis of the comparison between measured and calculated trends [36].

Fourteen participants coming from thirteen organizations and eleven countries have participated to the Phase II of BEMUSE program, submitting the required calculations and using seven thermal-hydraulic system codes (different versions of the same code have been used for the same case).

Main achievements of the Phase II, to be considered in the following phases of BEMUSE, are summarized as follows:

1) Almost all performed calculations appear qualified against the fixed criteria: few mismatches between results and acceptability thresholds have been characterized;
2) Dispersion bands of results appear substantially less than in ISP-13 (even though a lower number of participants submitted calculation in BEMUSE with respect to ISP-13): this testifies of code improvements in the last 20 years but especially in techniques for performing analysis.

4.5 The Internal Assessment of Uncertainty: The CIAU Code

The idea of Internal Assessment of Uncertainty came out in 1996 (see below) and was realized by the CIAU method that utilizes the basic approach of the UMAE at the UPISA [6]. This idea was triggered by the fact that all of the uncertainty methodologies suffer of two main limitations:

- The resources needed for their application may prove to be prohibitive, ranging to up to several man-years;
- The achieved results may be strongly methodology user dependent.

The last item should be considered together with the code-user effect [4], and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty methodology. Therefore, the Internal Assessment of Uncertainty (IAU) was requested as the follow-up of an International Conference, US NRC and OECD/CSNI held in Annapolis in 1996.

The CIAU approach here considered [6], has been developed having in mind the objective of removing the above limitations. Definitely, the "internal assessment of uncertainty" constitutes a desirable capability for thermal-hydraulic system codes allowing the "automatic" achievement of uncertainty bands associated with any code-calculation result. A description of the CIAU method is given in ref. [6], [37], [38], and [39].

The idea at the basis of the CIAU can be summarized by three items:

1) Build-up of NPP status: each status is characterized by the value of six relevant quantities (or phases) and by the value of the time since the transient start. Each of the relevant quantities is subdivided into a suitable number of intervals that may be seen as the edges of hypercubes in the phase-space. The duration of the transient scenario is also sub-divided into intervals.
2) Association of uncertainty with NPP status: accuracy values derived from the analysis of experimental data are associated to each NPP status.
3) Use of the method: at each time, the CIAU code-calculation result is associated to a time interval and to a hypercube, i.e. a NPP status, from which the uncertainty values are taken and associated with the current value of the prediction.

A simplified flow diagram of the CIAU is given in Fig.5, where two main parts can be seen. The former deals with the development of the method and the latter with its application. The CIAU development took benefit from the experience gained in the development of the UMAE uncertainty methodology [5].

The development of the method implies the availability of qualified experimental data (block a in Fig.5), of qualified system codes calculation results (block b), of postulated

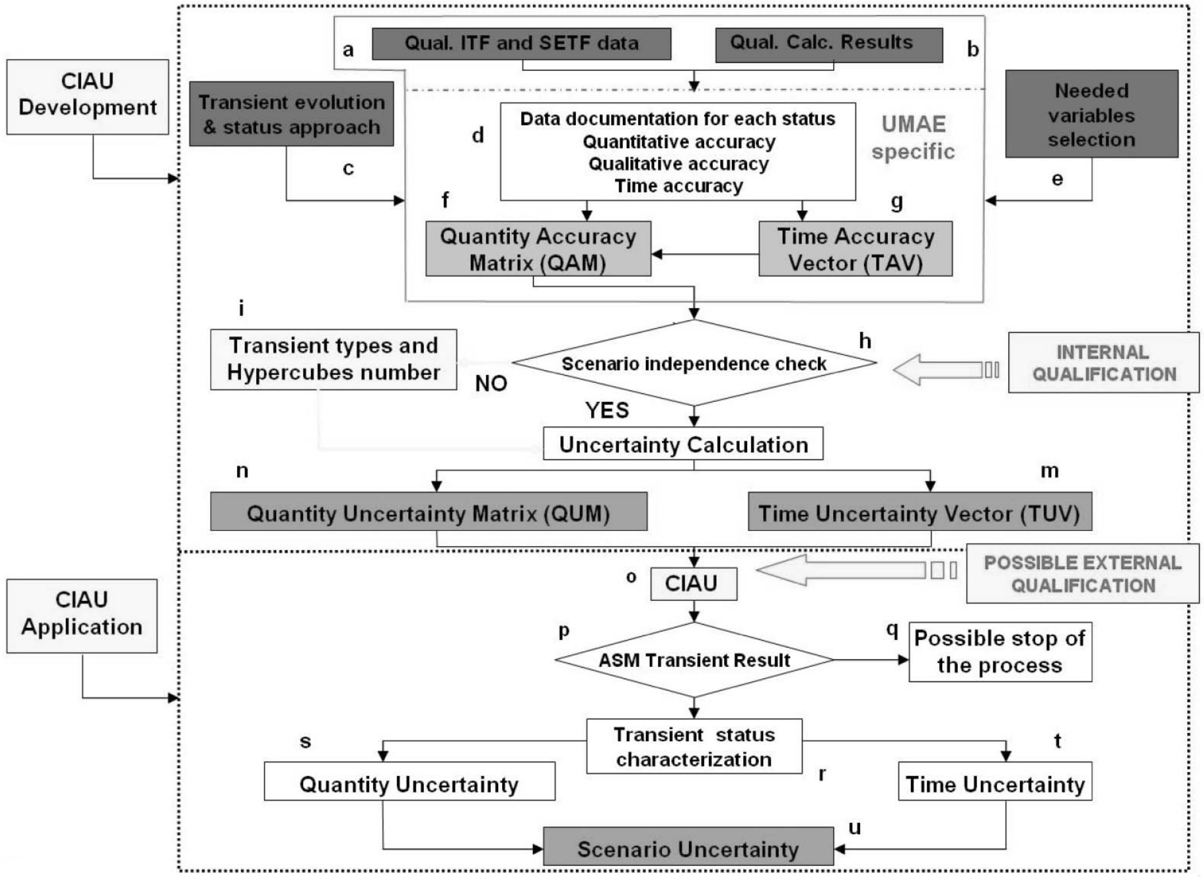

Fig. 5. Simplified Flow Diagram of the CIAU

transients including the definition of plant status (block c) and the selection of variables in relation to which the uncertainty must be calculated (block e). The support of experimental data (block a) is considered mandatory whatever is the qualification process. Qualified code results (block b) signify the run of qualified code in a qualified computer/compiler, by a qualified user using a qualified nodalization [1]. The qualification level of the code results is evaluated from a qualitative and a quantitative point of view, making use of the FFTBM.

The CIAU method allows getting "automatic" achievement of uncertainty bands associated with any code-calculation result. These errors bands are generated from a built-up database including former simulation errors with respect to relevant experimental data. These errors constitute a combination of several sources of uncertainties as for instance, to the code modeling approximations, the user effects, the nodalization approaches, as well as the measurement accuracy. To each hypercube is associated a quantitative accuracy which has been calculated from the simulation of al large set of experimental transients performed in test facilities of NPPs (Table 7). The combination of accuracy values coming from uncertainties issued from the aforementioned tests-calculations allows the derivation of continuous error bands enveloping any time dependent variables that are the output of a system code calculation.

The hypercube is identified by six driving quantities. In the case of pressurized water nuclear reactor transients those quantities are the primary system pressure, the steam generator system pressure, the mass inventory in primary system, the core power, the rod surface temperature at 2/3 of core height, and the steam generator down-comer level.

A consistent ensemble of uncertainty values is included in any set of hypercubes. Each hypercube is determined by the six above selected variables that are representative of a generic transient scenario. The error bands are obtained from the extrapolation of hypercubes resulting from the aforementioned database. Therefore, the size of the discrepancies is governed by the number of experiment-calculations included in the database; the larger is the database size the

Table 7. Comparison Among Relevant Features of Uncertainty Methods

	FACILITY or PLANT	TEST	TRANSIENT	
			Type	
21	RD-14m	B9401	LBLOCA	
22	LOBI/MOD2	BT-02	LOFW	
23	ANGRA-1	RES-11-99	Black-Out	Station black out
24	PKL	PKL-B	SBLOCA	
25	VVER-1000	MCP01	MCP	MCP Restart
26	VVER-440	MCP02	MCP	MCP Trip
27	VVER-440 BC - V-213	BC-V213-05	LBLOCA	70 mm break diameter and 'far' position
28	PANDA	ISP42-Ph-A	Containment Pressurization	Passive Containment Cooling System Start-Up
29	PANDA	ISP42-Ph-C	Containment Pressurization	Long-Term Passive Decay Heat Removal
30	LOFT	L2-3	LBLOCA	A_r = 200% of A_{max} in CL - 100% power Delayed coast down of primary coolant pumps
31	PSB	CL-05-03	SBLOCA with AM	A_r = 0,5% of A_{max} in CL Normal operation systems for water supply to primary side
32	PSB	CL-07-08	SBLOCA with AM	A_r = 0,7% of A_{max} in CL
33	PSB	PSB-TEST3	SBLOCA	Accident with opening and failure of pressurizer safety valve
34	PSB	PSH-14-04	PRISE	1.4% leakage from primary to secondary side
35	PSB	PSB-CT-SB	SBLOCA	6-inch cold leg break (loop#4). Scaling analysis for Russian WWER type reactor.
36	PSB	PSB-TEST11	IBLOCA	Rupture on Upper Plenum accumulator line A_r = 11% equivalent of A_{max} in CL
37	PSB	CL-07-11	SBLOCA with AM	A_r = 0,7% of A_{max} in CL Normal operation systems for water supply to primary side
38	PSB	CL-07-12	SBLOCA with AM	A_r = 0,7% of A_{max} in CL
39	PSB	PSB-NC	NC	Step-wise coolant inventory reduction natural circulation.

more accurate is the predicted error band. The parameters for which uncertainty is calculated are currently three; the primary system pressure, the rod surface temperature at 2/3 core height and the primary mass inventory. These quantities where chosen due to the relevance of such variables and is not mandatory in the structure of the method.

The main drawbacks of this approach are related to the fact that the origin of uncertainty does not appear from the code results (it is impossible to distinguish contributions to the output error bands) and the accuracy of the results are limited by the available error database size [38].

4.6 Application of Industrial Methods

Several applications of the CIAU methodology with relevance to the nuclear industry are presented hereafter.

4.6.1 KRSKO Case

The two-loop Westinghouse reactor of Krsko [650 MW (electric)] constitutes the reference NPP. The list of transients that have been calculated by the RELAP5/ MOD3.2 can be found in [6]. The main boundary conditions are also reported. The initial conditions correspond to the nominal conditions for the operation of the NPP. Large and small break LOCA and transients not involving the loss of integrity of the primary circuit are part of the list of the considered transients. No attention has been paid to constraints posed by the licensing process or to results of probabilistic studies when planning the sequences of imposed events or selecting quantities like break area and position. The results of the final step of the CIAU process, related to some of the transients in [6] are given in Fig.6 In all cases, the thick

TRANSIENT		Scaling Factor 1 / KV	N° Hyper-cubes Involved	End of Test for CIAU (s)
Secondary-Side Significant Condition	Emergency System in Primary Side			
		1/60	27	900
		1/712	15	9913
		1/1	7	178
		1/145	10	6000
		1/1	3	120
		1/1	3	599
		1/1000	10	160
------	------	1/40	1	5400
------	------	1/40	1	7000
------	Accumulator, HPIS and LPIS in cold leg	1/50	38	200
	Failure of HPIS and LPIS	1/300	21	2235
	Delayed accident management (analogous to BETHSY 9-1b)	1/300	30	3675
		1/300	6	2438
		1/300	83	12208
		1/300	30	2585
		1/300	31	925
	Failure of HPIS and LPIS	1/300	25	3565
	Failure of HPIS cool-down through SS and HPIS train in affected loop	1/300	23	3290
		1/300	97	21050

line is the result of the Analytical Simulation Model (ASM), and the thin lines bound the predicted uncertainty.

4.6.2 UMS Case

A Small Break LOCA (SBLOCA) experiment performed in the Japanese facility LSTF was selected as objective of the analysis of the international UMS group [34]. Thermalhydraulic system code calculations were performed in order to simulate the transient evolution and uncertainty was calculated. The experimental data were used to show the success of the uncertainty evaluation and, eventually, to qualify the uncertainty methodology.

The success of the application was the demonstration that the predicted uncertainty bands bound the experimental data. UMAE was one of the methodologies successfully

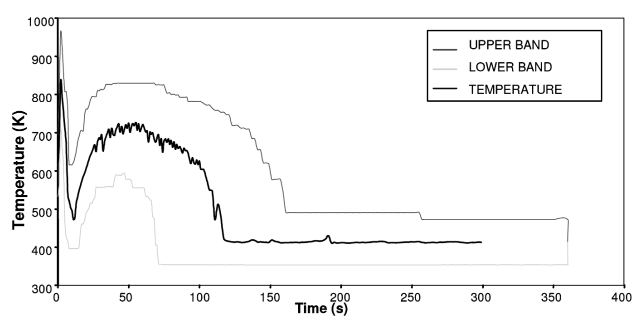

Fig. 6. Application of the CIAU to the Analysis of a Large Break LOCA in a Two-loop PWR: Rod Surface Temperature at 2/3 Core Height and Related Uncertainty Bands

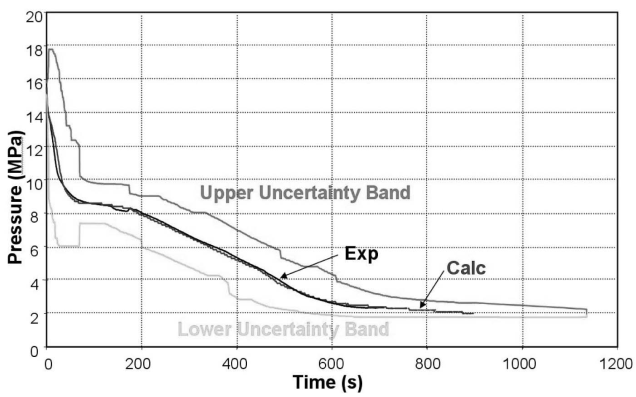

Fig. 7. Application of the CIAU to the UMS: Uncertainty Bands in Predicting Primary System Pressure

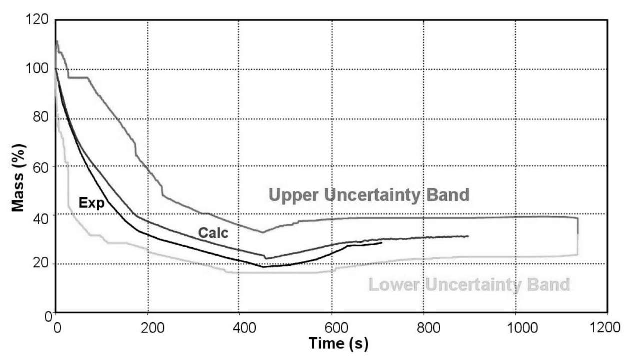

Fig. 8. Application of the CIAU to the UMS: Uncertainty Bands in Predicting Primary System Mass Inventory

applied in the process. RELAP5/2 and CATHARE2v1 .3u codes have been used for the prediction. The application of the CIAU to the same test has been completed adopting the RELAP5/3.2-beta version code. Typical results are given in Fig.7 to Fig.9. The best estimate code prediction, the experimental data and the upper and lower predicted uncertainty bands are reported.

4.6.3 Angra-2 Case

Two safety studies have been carried out by the CIAU method, that are relevant to the nuclear industry. Results, outlined hereafter, are derived from [39] and [40] where more details can be found. In the former study, the CIAU application aimed at performing an independent BEPU analysis of the LBLOCA-DBA of the Angra-2 PWR NPP. The analysis is classified as 'independent' in the sense that it was carried out by compu-tational tools (code and uncertainty method) different from those utilized by the applicant utility.

The main results are summarized in Fig.10, where Peak Cladding Temperature (PCT) and related uncertainty bands obtained by the CIAU and by the computational tools adopted by applicant, are given. The following comments apply:

- The CIAU (and the applicant) analysis has been carried out as best-estimate analysis: however, current rules for such analysis might not be free of undue conservatism and the use of peak factors for linear power is the most visible example.
- The conservatism included in the reference input deck constitutes the main reason for getting the 'PCT licensing' from the CIAU application above the acceptability limit of 1200 °C.
- The amplitude of the uncertainty bands is quite similar from CIAU and applicant. Discrepancies in the evaluation of 'PCT licensing' outcome from the way of considering

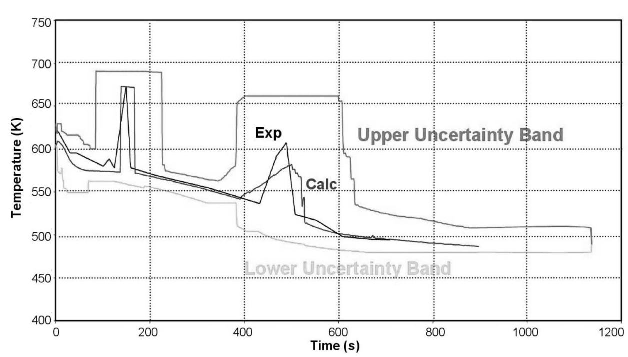

Fig. 9. Application of the CIAU to the UMS: Uncertainty Bands in Predicting Rod Surface Temperature at 2/3 Core Height

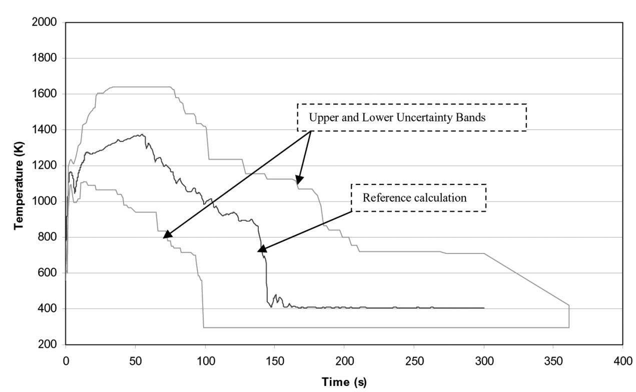

Fig. 10. Result of CIAU Application to Angra-2 LBLOCA Analysis: Uncertainty Bands for Rod Surface Temperature at 'Axial Level 9' of the Hot Rod Realistic, Obtained by the Reference Run

the 'center' of the uncertainty bands. In the case of CIAU, the 'center' of the uncertainty bands is represented by the phenomenological result for PCT obtained by the reference calculation (1100 °C in Fig.10). In the case of applicant the 'center' of the uncertainty bands is a statistical value obtained from a process where the reference calculation has a role (796 °C in Fig.11).

- The results of the CIAU method are supported by a number of 'finalized' sensitivity studies as large as about 150 (i.e. about 150 LBLOCA calculation have been performed to confirm the CIAU uncertainty results).
- The reference best estimate PCT calculated by the applicant (result on the left of the Fig.11) plus the calculated uncertainty is lower than the allowed licensing limit of 1473 K.
- The reference best estimate PCT calculated by CIAU (central result in the Fig.11) is higher than the PCT 'proposed' by the applicant and the upper limit for the rod surface temperature even overpasses the allowed licensing limit of 1473 K thus triggering licensing issues.

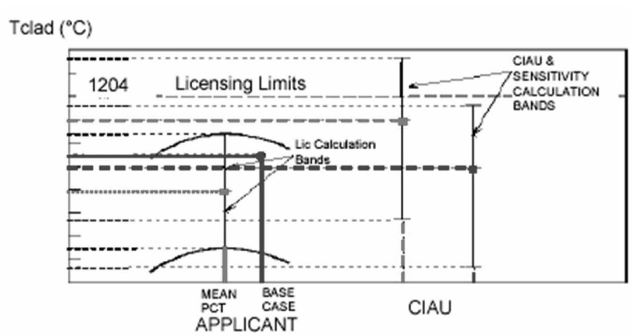

Fig. 11. Angra-2 LBLOCA Uncertainty Evaluation: Final Result from the CIAU Study and Comparison with Results of the Applicant

4.6.4 Best Estimate and Uncertainty Evaluation of LBLOCA 500 mm for Kozloduy-3

The analysis of the 'LBLOCA 500 mm' transient [40] was carried out by RELAP5/3.2 code. The specific purposes of the analysis include the assessment of the results and the execution of an independent safety analysis supported by uncertainty evaluation. A BE transient prediction of the 'LBLOCA 500 mm' was performed. Evaluation of the uncertainty was performed by CIAU for the RPV upper plenum pressure, the mass inventory in primary system and the hot rod cladding temperature. Only the last parameter is shown in Fig.12 together with the uncertainty bands. The most relevant result is the demonstration that the PCT in the concerned hot rod is below the licensing limit. In the same Fig.12, bounding results from two conservative calculations obtained by a BE code utilizing conservative input assumptions are given. One is the conservative calculation 'Driven' Conservatism (DC), the other is the conservative calculation performed by UPISA 'Rigorous' Conservatism (RC).

The following can be noted:

a) The conservative calculation DC is not "conservative" and does not bound entirely the BEPU upper bound. This implies that code uncertainties are not properly accounted for by the adopted conservative input parameter values.

b) The conservative calculation performed by [40] is correctly conservative, but its conservatism is such to cause PCT above the licensing limit. The comparison between the conservative PCT obtained by UPISA and the upper bound of the BEPU calculation shows the importance of using a full BE approach with a suitable evaluation of uncertainty.

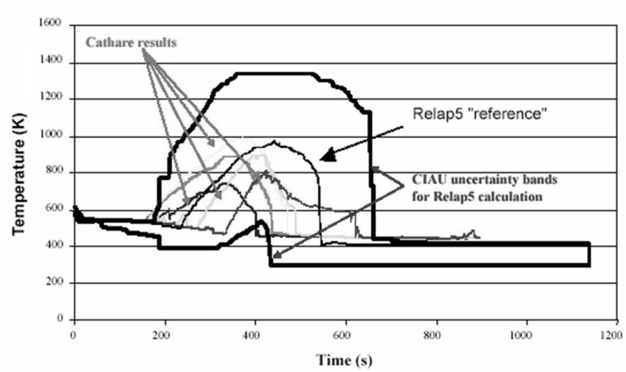

Fig. 12. Uncertainty Analysis of the '200 mm' LOCA-DBA of VVER-440 NPP: Main Result from CIAU Application

5. NEW TRENDS

In the light of the sustained development in computer technology, and the maturity of computational techniques as CFD and Computational Multi-Fluid Dynamics CMFD methods [41], the possibilities of code capabilities have been enlarged substantially. Nowadays, it becomes possible to switch to new generation of computational tools and perform advanced safety evaluations and design optimizations that were not possible few years ago. One of these techniques consists in coupling advanced codes to get more realistic simulations of complex phenomena and transients in Nuclear Power Plants (NPP). The application of such method is mandatory for [42];

1) Calculating in more detail situation where strong interaction between neutronics and thermal-hydraulics exist. This includes the coupling between 3D Neutron

kinetics and THSC codes. This is particularly true for the simulation of

- Almost all Reactivity Initiated Accidents (RIA) as:
  - -The reactivity increases induced by thermal-hydraulic effects.
  - -The asymmetric perturbation in the core.
  - -The re-criticality cases. This is particularly emphasized for high values of moderator temperature coefficient, for increased high burnup fuel, or for extended use of MOX fuel.
  - -The local boron dilution accident in PWR and VVER.
  - -All Anticipated Transients Without Scram (ATWS) and other Beyond DBA.
- The BWR stability issues in plant conditions and beyond the stability threshold [43].
- Nuclear Power improvement programs, which generate the demand for reducing uncertainties.

2) Calculating the system behaviour and local behaviour simultaneously. This includes:

- THSC / core TH (subchannel) coupled codes. The boundary conditions of each code are given at the boundaries of core region;
- TH / fuel behaviour coupled code. The boundary conditions of each code are given at the fuel surface;
- TH / CFD coupled code. The boundary conditions of CFD codes are provided by TH codes.

3) Calculating in more detail the interaction between thermal-hydraulic behaviour and mechanical behaviour. This includes:

- TH / Structure mechanics coupled code (e.g. effect on component vibration characteristics of fluid dynamics).
- TH / containment behaviour coupled code (e.g. to calculate primary system thermal-hydraulic behaviour and containment behaviour simultaneously after LOCA).

New computational tools constitute a challenge for the uncertainties assessment and address further development of the uncertainty methodologies toward the introduction of such a capabilities for the coupled codes. CIAU-TN constitutes a pioneering effort in this field, realizing the coupling between the UMAE uncertainty methodology and the RELAP5/PARCS coupled code [7]. The demonstration of the feasibility of others approaches has been published in [33] and [44] Notwithstanding that the full implementation and use of the procedure requires a larger assessment effort, the results obtained give an idea of the errors expected from the application of the present computational tool to problems of practical interest.

6. 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. However any calculation from a best estimate code, to be meaningful, needs an uncertainty evaluation. This is valid for various applications of the codes ranging from licensing studies to training of plant operators. In carrying out such activities, problems like qualitative and quantitative accuracy evaluation, philosophical basis for code assessment, user effect, nodalization qualification, computer/compiler effect, scaling up of calculation results and of code capabilities, influence of boundary conditions on calculation results, have been addressed and a solution has been proposed or an answer has been given.

Most of the problems and questions that arose 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. It is clear that funding for performing expensive research campaigns that was justified for ensuring the safety of existing reactors, will no longer be available. The following areas can be mentioned where experimental data will have an important role:

- Use of BE in licensing of new NPP including advanced reactors, licensing renewal, life prolongation and power upgrading of existing reactors.
- Safety analysis of NPP that were not been the main target of the scientific community in the past. This includes VVER, RMBK and CANDU types reactors.
- Qualification of suitable single phase CFD codes with system codes.
- Development of suitable two-phase CFD or CMFD codes. The program proposed by NRC in relation to the tracking of the interfacial area, can be considered in this connection [45].

The international community should concentrate its efforts on spreading the experience gained by relevant organizations, also reaching agreements in relation to specific issues.

We conclude that 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. However, new scientific goals must be achieved. To this aim, the following requirements and recommendations have been identified:

- A general-purpose system code should be developed essentially including multifluid capability and "open" interfaces for an easy coupling with other codes in areas like neutronics (for implementing presently available 3-D codes), CFD, structural mechanics (e.g. for Pressurized Thermal Shock studies), and containment.
- User interfaces: although a well established set of correlations and models has to be the basis for the system code,

it is important to add an efficient user interface to allow additional model/components to be tested. In this same area, boundary volumes and junctions as well as heat structures and trips should be opened to external access to gain code flexibility, needed for an application range not limited to the nuclear field;
- A continuous release of new code versions is not of benefit for the user, e.g., user choices (or experience) valid for a specific version may be not valid for the new one: new code versions should be released only after thorough assessment carded out by a well identified group of users ;
- Detailed requirements for user qualification should be developed;
- Current computer capabilities should be properly used;
- The full exploitation of "advanced" system codes implies their acceptability by the licensing authorities. This requires the characterization of uncertainty methods that should not use engineering judgement in the application phase. Conditions should be made clear for accepting the available uncertainty methods in the licensing process.

ACRONYMS

ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design Basis Accident
BE	Best-Estimate
BEPU	Best-Estimate Plus Uncertainty
BWR	Boiling Water Reactor
CFD	Computational Fluid-Dynamics
CIAU	Code with capability of Internal Assessment of Uncertainty
CPU	Central Process Unit
CSNI	Committee on the Safety of Nuclear Installations
DBA	Design Basis Accident
EC	European Commission
GRS	Gesellschaft fuer Anlagen- und Reaktorsicherheit
IAEA	International Atomic Energy Agency
ITF	Integral Test Facility
LOCA	Loss Of Coolant Accident
LBLOCA	Large Break Loss Of Coolant Accident
LWR	Light Water Reactor
MOX	Mixed U-Pu oxide nuclear fuel
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
OECD	Organization for Economic Co-operation and Development
PCT	Peack Cald Temperature
PWR	Pressurized Water Reactor
RIA	Reactivity Initiated (or Induced) Accident
TH	Thermal-Hydraulic
THSC	Thermal-Hydraulic System Code
UPISA	University of Pisa
VVER	Water-cooled Water-moderated Energy Reactor
3D or 3-D	Three-Dimensional

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