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ERROR SOURCES CONSIDERED IN THE "UMAE DRIVEN" CIAU METHODOLOGY

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Abstract

Uncertainty analysis aims at characterizing the errors associated with experimental data and with results of computer codes when applied to the predictions of nuclear power plants related scenarios, in contradistinction with sensitivity analysis, which aims at determining the 'amount of change' in the predictions of codes when one or more input parameters varies within proper ranges of interest.

An outline of the features of independent approaches for estimating the uncertainty associated with predictions of system thermal-hydraulic codes is provided. The approach, according to which the uncertainty derives from the comparison between relevant measured data and results of corresponding code calculations, is discussed in details.

Within the framework of this approach, the UMAE (Uncertainty Method based on the Accuracy Extrapolation) method has been developed first and, later on, used as virtual engine for the method named Code with capability of Internal Assessment of Uncertainty (CIAU). The principle of UMAE is the consideration of the measure of accuracy as a statistical quantity independent of the scale. On this basis the value of accuracy measured at different scales is also valid at the scale of the Nuclear Power Plant and constitutes the uncertainty.

Introduction

Uncertainty analysis aims at characterizing the errors associated with experiments and predictions of computer codes, in contradistinction with sensitivity analysis, which aims at determining the rate of change (i.e., derivative) in the predictions of codes when one or more (typically uncertain) input parameters varies within its range of interest.

The first approach, reviewed as the prototype for propagation of code input uncertainties includes the "CSAU method" (Code Scaling, Applicability and Uncertainty) and the majority of methods adopted by the nuclear industry such as the so-called "GRS method". Although the entire set of the actual number of input parameters for a typical NPP (Nuclear Power Plant) input deck, ranging up to about 105 input parameters, could theoretically be considered as uncertainty sources by these methods, only a 'manageable' number (of the order of several tens) is actually taken into account in practice. Ranges of variations, together with suitable

PDF (Probability Density Function) are then assigned for each of the uncertain input parameter actually considered in the analysis. The number of computations using the code under investigation needed for obtaining the desired confidence in the results can be determined theoretically (it is of the order of 100). Subsequently, an additional number of computations (ca. 100) with the code are performed to propagate the uncertainties inside the code, from inputs to outputs (results).

The focus in this paper is given to the second approach, which based on the propagation of code output errors, as representatively illustrated by the UMAE-CIAU (Uncertainty Method based upon Accuracy Extrapolation 'embedded' into the Code with capability of Internal Assessment of Uncertainty). Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of 'relevant' experimental data, here, the word 'relevant' is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically Integral Test Facility (ITF) data, and assuming the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions, then the resulting (error) database is processed and the 'extrapolation' of the error takes place. Conditions for the extrapolation are:

- building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations;

- performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

Additionally, a third approach described in this paper, which is based on ASAP (Adjoint Sensitivity Analysis Procedure) and GASAP (Global Adjoint Sensitivity Analysis Procedure) methods extended to performing uncertainty evaluation in conjunction with concepts from Data Adjustment and Assimilation (DAA). The ASAP is the most efficient deterministic method for computing local sensitivities of large-scale systems, when the number of parameters and/or parameter variations exceeds the number of responses of interest. The DAA is the technique by which experimental observations are combined with code predictions and their respective errors to provide an improved estimate of the system state; in other words, DAA uses dynamic models to extract information from observations in order to reconstruct the structure of the system and reduce uncertainties in both the system parameters and responses. The reason for considering this approach derives from its potential to open an independent way (i.e. different from propagation of code input errors or from propagation of code output errors) for performing global uncertainty analysis.

1. Background and objectives

Let's consider three relevant definitions, i.e., in alphabetic order, accuracy, sensitivity and uncertainty, as they are commonly accepted in the sector of deterministic accident analysis within the more general framework of nuclear reactor safety technology.

Accuracy is defined, [1], as "the known bias between a code prediction and the actual transient performance of a real facility". Therefore, the evaluation of accuracy implies the availability of a calculation result and of a measured value. Point values and continuous time trends shall be included in the definition. The experimental error is not part of the definition. However, in the majority of cases of practical interest in the area of accident analysis of nuclear power plants, the error that characterizes the measurement is much lower of the error (i.e. the accuracy) that characterizes the comparison between measured and predicted values.

The sensitivity is, according to [2], "… the study of how the variation in the output of a model (numerical or otherwise) can be apportioned, qualitatively or quantitatively, to different sources of variation, and of how the given model depends upon the information fed into it.". Furthermore, "Sensitivity analysis studies the relationships between information flowing in and out of the model.". These definitions imply that the parameter values that characterize both (and only) the boundary and initial conditions, e.g. representative of a system, and the numerical structure of a correlation embedded into the model (or code) constitute the typical objective of a Sensitivity Analysis (SA).

The uncertainty is the unknown error that characterizes the prediction of any code or model. The uncertainty analysis is, according to [1] and related to system thermal-hydraulic code predictions, "an analysis to estimate the uncertainties and error bounds of the quantities involved in, and the results from, the solution of a problem. Estimation of individual modeling or overall code uncertainties, representation (i.e. nodalization related) uncertainties, numerical inadequacies, user effects, computer compiler effects and plant data uncertainties for the analysis of an individual event". Furthermore, to conclude with a citation from [2], "… uncertainty is not an accident of the scientific method but its substance". Within the present context, the uncertainty is the necessary supplement for a best-estimate thermal-hydraulic code prediction; see also [3].

The reason why an accuracy analysis (AA) is performed is mainly connected in the sector under investigation here (i.e. the deterministic accident analysis) with the demonstration of qualification for computer codes. The accuracy analysis implies the availability of relevant experimental data and of tools to characterize the resulting discrepancies from qualitative and quantitative viewpoints, e.g. [4] and [5].

The reasons why a sensitivity analysis (SA) is performed are strongly affected by the type and the objectives of the model and may range from verification purposes, to finding singular points (e.g. maximum and minimum) of an assigned output quantity, or the factors that mostly contribute to that output, or the correlation among input variables. It can be premised that needs for SA come from the fundamental principles of quality assurance.

The reasons why an uncertainty analysis (UA) is performed come from nuclear safety principles and primarily from concepts like defense-in-depth. It must be ensured that the nominal result of a code prediction, 'best-estimate' in the present case, is supplemented by the uncertainty statement, that can be simplified as 'uncertainty bands', in such a way that connected safety margins are properly estimated.

The key result from AA is the demonstration of the qualification level of a code and the characterization of the range of parameters over which the code should be considered as qualified and applicable to situation of interest to nuclear reactor safety. The AA should also provide an answer to the scaling issue, [6].

The key result from SA is the influence of input parameters upon selected output quantities and the evaluation of the relative influence of input parameters, according to the definition given above.

The key results from UA are error bands that bound the best-estimate predictions. Point value error bands can be distinguished from continuous error bands that bound one or several curves, as well as from three-dimensional graphic representations where instantaneous values for quantity-error (e.g. pressure) are reported together with time-error as a function of time, [7].

Therefore AA, SA and UA are closely linked, but important differences can be identified. All that is needed for a meaningful SA is the model and the input values, while UA attempts to estimate the actual error band value for an output; as a consequence, it needs a reference value typically not available (thus the definition of 'unknown' error). AA, on the other hand needs relevant experimental data. As an example, the check that an assigned model satisfies the first or the second principle of thermodynamics may not be the objective of SA, but it is the objective for UA and can be confirmed following AA. Furthermore, when performing SA, the values of the concerned input parameters are varied arbitrarily around the initial (or nominal) value to a 'small' or to a 'large' extent depending upon the scope of the analysis; when performing the UA, whatever is the method adopted, a range of variation for the concerned input parameters must be assigned or available. SA may be a way to perform UA if input parameters are properly selected with proper ranges of variation.

The present paper focuses on UA. The historical triggering for UA in the area of nuclear reactor thermal-hydraulics may be traced as the Regulatory Guides (e.g. RG 1-157 and, more recently 1-203, [3]) issued by US NRC to streamline the application of codes when demonstrating the compliance of reactor accident scenario calculation with the criteria in Appendix K of 10 CFR 50-46. However, an international code assessment project conducted within OECD/NEA/CSNI since the beginning of eighties also showed the exigency for UA.

In the meantime, a number of uncertainty methodologies were proposed in other countries, including the GRS, the UMAE and the AEA Technology methods, as summarized in [9] and [10]. These methods use different techniques and procedures to obtain the uncertainties on key calculated quantities. Presently, uncertainty bands can be derived (both upper and lower) for any desired quantity throughout the transient of interest, not only point values like peak cladding temperature. For one case, the uncertainty method is coupled with the thermal-hydraulic code and is denominated CIAU (Code with capability of Internal Assessment of Uncertainty, [11]) and discussed below in more detail. All these methods are described into detail in [12], including examples of applications to cases of industrial interest.

The purpose of the present paper is twofold: a) to outline the sources of errors/uncertainty in the results of the system thermal-hydraulic codes and how they are addressed in the UMAE driven methodology; b) to identify the roadmaps for uncertainty evaluation adopted by the methods currently applied to the cases of industrial interest, making reference to the classification proposed in [12].

2. The approaches to calculate the uncertainty

The features of independent approaches for estimating uncertainties are reviewed below.

The propagation of code input errors (Fig. 1): this can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code.

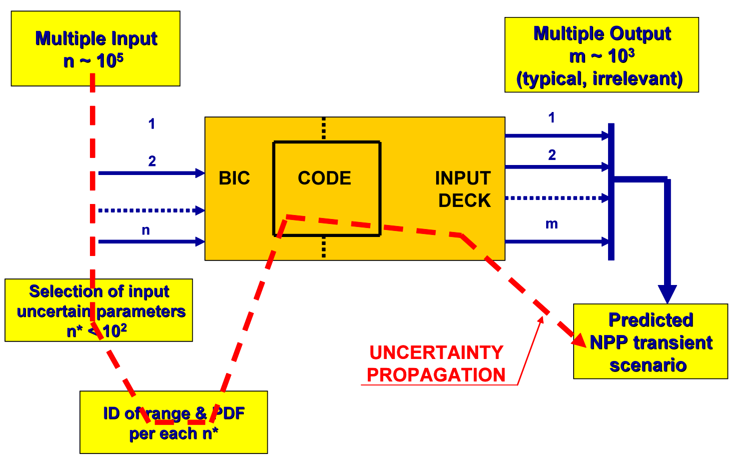

Figure 1 Uncertainty methods based upon propagation of input uncertainties (GRS method)

The propagation of code output errors (Fig. 2): this is the only demonstrated independent working alternative to the previous one and has also been used for industrial applications. It makes full and direct reference to the experimental data and to the results from the assessment process to derive uncertainty. In this case the uncertainty prediction is not propagated throughout the code.

The 'third' approach, (Fig. 3): this is an independent way, i.e. different from propagation of code input errors or from propagation of code output errors is based on Adjoint Sensitivity Analysis Procedure (ASAP), Global Adjoint Sensitivity Analysis Procedure (GASAP), [13] and [14] and Data Adjustment/Assimilation (DAA) methodology [15] by which experimental and calculated data, including the computation of sensitivities (derived from ASAP), are mathematically combined for the prediction of the uncertainty scenarios. The approach is reviewed hereafter as a deterministic method.

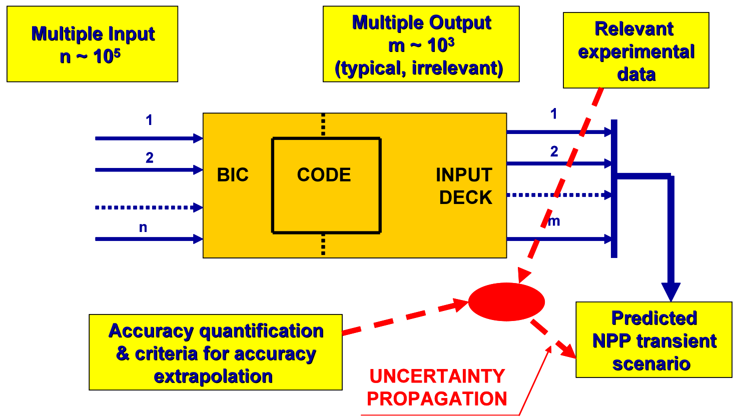

Figure 2 Uncertainty methods based upon propagation of output uncertainties (CIAU method)

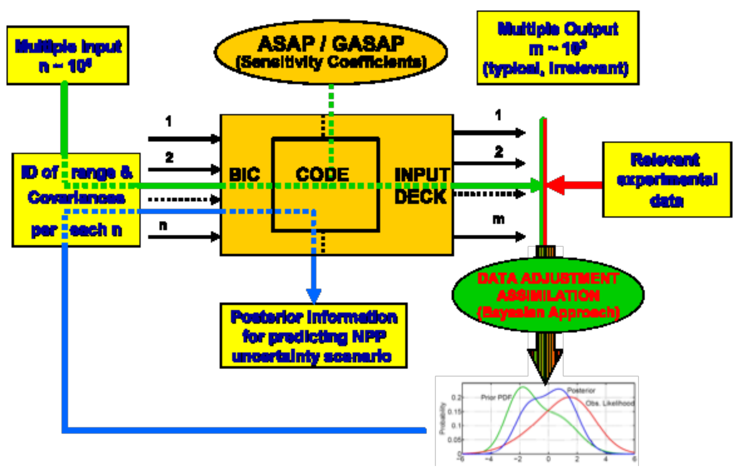

Figure 3 Uncertainty methodology based on Adjoint Sensitivity Analysis Procedure and Data Adjustment/Assimilation

The first approach, reviewed as the prototype for propagation of code input errors, is the so-called "GRS method" [16], which includes the so-called "CSAU method" (Code Scaling, Applicability and Uncertainty) [8] and the majority of methods adopted by the nuclear industry. Although the entire set of the actual number of input parameters for a typical NPP (Nuclear Power Plant) input deck, ranging up to about 105 input parameters, could theoretically be considered as uncertainty sources by these methods, only a 'manageable' number (of the order of several tens) is actually taken into account in practice. Ranges of variations, together with suitable PDF (Probability Density Function) are then assigned for each of the uncertain input parameter actually considered in the analysis.

The number of computations needed for obtaining the desired confidence in the results can be determined theoretically by the Wilks formula [17]. Subsequently, the identified computations (ca. 100) are performed using the code under investigation to propagate the uncertainties inside the code, from inputs to outputs (results). The logical steps of the approach are depicted in Fig. 1.

The main drawbacks of such methods are connected with: a) the need of engineering judgment for limiting (in any case) the number of the input uncertain parameters; b) the need of engineering judgment for fixing the range of variation and the PDF for each input uncertain parameter; c) the use of the code-nodalization for propagating the uncertainties: if the code-nodalization is wrong, not only the reference results are wrong but also the results of the uncertainty calculations; d) the process of selecting the (about) 100 code runs is demonstrably not convergent, and the investigation of results from two or more different sets of 100 calculations shows different values for uncertainty. A study reported by KAERI in the framework of the Phase III of BEMUSE project [18] by performing a direct Monte-Carlo simulation consisting of 3500 runs simulating the Large Break Loss Of Coolant Accident (LBLOCA) L2-5 in the LOFT facility and comparing with samples of n = 59 and n = 93 calculations, resulted with the following considerations:

- From about 1000 runs, the mean value (equal to 1034 K) and the 95% empirical quantile (equal to 1173 K) of the first PCT (Peak Cladding Temperature) are almost stabilized;

- The 95% quantile value of 1173 K has to be compared with the value of 1219 K obtained with the sample of 93 calculations used for evaluating the upper tolerance limit of the first PCT in the BEMUSE project. A difference of 46 K has been attained;

- The dispersion of the upper limit obtained by using Wilks' formula at the first (i.e. the maximum value is retained) and second order (i.e. the second maximum value is retained), with a probability of 95% and a confidence level of 95%, was studied. The following aspects have to be outlined:

  - The spread of the results predicted for the upper limit of the first PCT is equal to roughly 200 K at the first order and 120 K at the second order;

  - At first order, among the 58 calculations, ranging from 1170 K to 1360 K, no-one was found significantly lower than the 95% quantile of the 3500 code runs, notwithstanding statistically 3 cases (i.e. 5% of 58) are expected;

  - At the second order, among 37 calculations, ranging from 1150 K to 1270 K, 1 case was found below 1173 K.

The second approach, reviewed as the propagation of code output errors, is representatively illustrated by the UMAE-CIAU (Uncertainty Method based upon Accuracy Extrapolation [19] 'embedded' into the Code with capability of Internal Assessment of Uncertainty [11, 7]). Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of 'relevant' experimental data, where here the word 'relevant' is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically Integral Test Facility (ITF) data, and assuming the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions [19], then the resulting (error) database is processed and the 'extrapolation' of the error takes place. Relevant conditions for the extrapolation are:

- Building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations;

- Performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

The main drawbacks of this method are as follows: (i) the method is not applicable in the absence of relevant experimental information; (ii) a considerable amount of resources is needed to establish a suitable error database, but this is a one-time effort, independent of subsequent applications of this method; (iii) the process of combining errors originating from different sources (e. g, stemming from different ITF or SETF (Separate Effect Test Facility), different but consistent nodalizations, different types of transient scenarios) is not based upon fundamental principles and requires detailed validation.

The third approach, depicted in Fig. 3, is based upon the powerful mathematical tools of ASAP, GASAP and DAA by which all parameters $\boldsymbol{\alpha}$ that affect any prediction, being part of either the code models or the input deck can be considered. The Adjoint Sensitivity Analysis Procedure (ASAP) [13, 14] is the most efficient deterministic method for computing local sensitivities $\mathbf{S}$ of large-scale systems, when the number of parameters and/or parameter variations exceeds the number of responses $\mathbf{R}$ of interest (that is the case of most problems of practical interest). In addition, also system's critical points y (i.e. bifurcations, turning points, saddle points, response extrema) can be considered and determined by the Global Adjoint Sensitivity Analysis Procedure (GASAP) [13, 14] in the combined phase-space formed by the parameters, forward state variables, and adjoint variables. Subsequently the local sensitivities of the responses R located at critical points y are analyzed by the ASAP.

Once the sensitivity matrix $\mathbf{S}$ of the responses $\mathbf{R}$ respect to the parameters α is available, the moment propagation equation is adopted to obtain the computed covariance matrix $\mathbf{C_R}$ of the responses starting from the covariance matrix $\mathbf{C}_{\boldsymbol{\alpha}}$ of the system parameters. The elements of the matrix $\mathbf{C}_{\boldsymbol{\alpha}}$ reflect the state of knowledge about the input (uncertainty) parameters that can be characterized by ranges and PDF. It is very well known that in system thermal-hydraulics only few elements of $\mathbf{C}_{\boldsymbol{\alpha}}$ are obtained from experimental observations (mainly from SETF), whereas for the major part of them engineering judgment is adopted for deriving ('first') guess values of ranges and PDF. The imperfect knowledge of the input uncertainty parameter obviously affects the computed responses $\mathbf{R}$ and the relative covariance $\mathbf{C_R}$ and constitutes the main reason for which proper experimental data (i.e. connected with the specific NPP transient scenario under investigation for uncertainty evaluation) are needed. The technique by which experimental observations are combined with code predictions and their respective errors to provide an improved estimate of the system state is known as Data Adjustment and Assimilation (DAA) and it is based on a Bayesian inference process.

The main drawbacks of this approach are as follows: (i) the method is not applicable in the absence of relevant experimental information; (ii) the adjoint model, needed for computing the sensitivity **S**, requires relatively modest additional resources to develop and implement if this is done simultaneously with the development of the original code; however if the adjoint model is constructed a posteriori, considerable skills may be required for its successful development and implementation; (iii) a considerable amount of resources is needed to establish a suitable database of improved estimates for the input parameters ($\boldsymbol{\alpha}^{\mathbf{IE}}$) and for the respective input covariance matrix ($\mathbf{C}_{\boldsymbol{\alpha}}^{\mathbf{IE}}$), but this is a one-time effort, independent of subsequent applications of the method.

The maturity of the methods at the first two bullets may be considered as proved also based upon applications completed within the framework of initiatives of international institutions (OECD/NEA [9, 18] and IAEA [1]). The reason for the consideration of the approach at the third bullet derives from its potential to open an independent way (i.e. different from propagation of code input errors or from propagation of code output errors) for performing global uncertainty analysis. In this case, the method itself, as an uncertainty procedure, is not an established technology, but it constitutes an established idea and framework to pursue a mathematically based road to evaluate the uncertainty in system code predictions. In the following sections, short descriptions of the most known methods belonging to the first two discussed approaches are given.

3. The UMAE-CIAU Methodology

3.1. Sources of Errors

Application of best-estimate (realistic) computer codes to the safety analysis of nuclear power plants implies the evaluation of uncertainties. This is connected with the (imperfect) nature of the codes and of the process of codes application. In other words, ‘sources of errors’ or ‘sources of uncertainty’ affect the predictions by best-estimate codes and must be taken into account. Three major sources of error are mentioned in the Annex II of the IAEA guidance Accident Analyses for Nuclear Power Plants, ref. [2]:

- Code or model uncertainty.
- Representation or ‘simulation uncertainty’.
- Plant uncertainty.

A more detailed list of uncertainty includes the following items:

A) Balance (or conservation) equations are approximate:

   - not all the interactions between steam and liquid are included,
   - the equations are solved within cylindrical pipes: no consideration of geometric discontinuities, situation not common for code applications to the analysis of Nuclear Power Plants transient scenarios;

B) Presence of different fields of the same phase: e.g. liquid droplets and film. Only one velocity per phase is considered by codes, thus causing another source or uncertainty.

C) Geometry averaging at a cross section scale: the need "to average" the fluid conditions at the geometry level makes necessary the 'porous media approach'. Velocity profiles happen in the reality: These correspond to the 'open media approach'. The lack of consideration of the velocity profile, i.e. cross-section averaging, constitutes an uncertainty source of 'geometric origin'.

D) Geometry averaging at a volume scale: only one velocity vector (each phase) is associated with a hydraulic mesh along its axis. Different velocity vectors may occur in the reality (e.g. inside lower plenum of a typical reactor pressure vessel, at the connection between cold leg and down-comer, etc.). The volume-averaging constitutes a further uncertainty source of 'geometric origin'.

E) Presence of large and small vortex or eddy. Energy and momentum dissipation associated with vortices are not directly accounted for in the equations at the basis of the codes, thus introducing a specific uncertainty source. In addition, a large vortex may determine the overall system behaviour (e.g. two-phase natural circulation between hot and cold fuel bundles), not necessarily consistent with the prediction of a code-discretized model.

F) The 2nd principle of thermodynamics is not necessarily fulfilled by codes. Irreversible processes occur as a consequence of accident in nuclear reactor systems. This causes 'energy' degradation, i.e. transformation of kinetic energy into heat. The amount of the transformation of energy is not necessarily within the capabilities of current codes, thus constituting a further specific energy source.

G) Models of current interest for thermal-hydraulic system codes are constituted by a set of partial derivatives equations. The numerical solution is approximate, therefore, approximate equations are solved by approximate numerical methods. The 'amount' of approximation is not documented and constitutes a specific source of uncertainty.

H) Extensive and unavoidable use is made of empirical correlations. These are needed 'to close' the balance equations and are also reported as 'constitutive equations' or 'closure relationships'. Typical situations are:

- The ranges of validity are not fully specified. For instance, pressure and flowrate ranges are assigned, but void fraction, or velocity (or slip ratio) ranges may not be specified.
- Relationships are used outside their range of validation. Once implemented into the code, the correlations are applied to situations, where, for instance, geometric dimensions are different from the dimensions of the test facilities at the basis of the derivation of the correlation. One example is given by the wall-to-fluid friction in the piping connected with reactor pressure vessel: no facility has been used to derive (or to qualify) friction factors in two phase conditions when pipe diameters are of the order of one meter. In addition, once the correlations are implemented into the code, no (automatic) action is taken to check whether the boundaries of validity, i.e. the assigned ones, are over-passed during a specific application.
- Correlations are implemented approximately into the code. The correlations, apart from special cases, are derived by scientists or in laboratories that are not necessarily

aware of the characteristics or of the structure of the system code where the correlations are implemented. Furthermore, unacceptable numeric discontinuities may be part of the original correlation structure. Thus, correlations are 'manipulated' (e.g. extrapolated in some cases) by code developers with consequences not always ascertained.

- Reference database is affected by scatter and errors. Correlations are derived from ensembles of experimental data that unavoidably show 'scatter' and are affected by errors or uncertainties. The experimentalist must interpret those data and achieve an 'average-satisfactory' formulation.

I) A paradox: shall be noted: 'Steady State' & 'Fully Developed' (SS & FD) flow condition is a necessary prerequisite or condition adopted when deriving correlations. In other terms, all qualified correlations must be derived under SS & FD flow conditions. However, almost in no region of the Nuclear Power Plant those conditions apply during the course of an accident.

J) The state and the material properties are approximate. Various materials used in a NPP are considered in the input deck, including liquids, gases and solids. Thermo-physical properties are part of the codes or constitute specific code user input data. These are of empirical nature and typically subjected to the limitations discussed under item H). A specific problem within the current context can be associated with the derivatives of the water properties.

K) Code User Effect (UE) exists. Different groups of users having available the same code and the same information for modelling a Nuclear Power Plant do not achieve the same results. UE (see also below) is originated by:

- Nodalization development, see also item N), below.
- Interpreting the supplied (or the available) information, usually incomplete, see also item M) below;
- Accepting the steady state performance of the nodalization;
- Interpreting transient results, planning and performing sensitivity studies, modifying the nodalization and finally achieving "a reference" or "an acceptable" solution;

The UE might result in the largest contribution to the uncertainty and is connected with user expertise, quality and comprehensiveness of the code-user manual and of the database available for performing the analysis.

L) Computer/compiler effect exists. A computer code is developed making use of the hardware selected by the code developers and available at the time when the code development starts. A code development process may last a dozen years during which period profound code hardware changes occur. Furthermore, the code is used on different computational platforms and the current experience is that the same code with the same input deck applied within two computational platforms produces different results. Differences are typically small in 'smoothly running transients', but may become noticeable in the case of threshold- or bifurcation-driven transients.

M) Nodalization (N) effect exists. The N is the result of a wide range brainstorming process where user expertise, computer power and code manual play a role. There is a number of required code input values that cannot be covered by logical recommendations: the user expertise needed to fix those input values may reveal inadequate and constitutes the origin of a specific source of uncertainty.

N) Imperfect knowledge of Boundary and Initial Conditions (BIC). Some BIC values are unknown or known with approximation: the code user must add information. This process unavoidably causes an impact on the results that is not easily traceable and constitutes a specific source of uncertainty.

O) Code/model deficiencies cannot be excluded. The system code development started toward the end of the sixties and systematic assessment procedures were available since the eighties. A number of modelling errors and inadequacies have been corrected or dealt with and substantial progress has been made in improving the overall code capabilities. Nevertheless, deficiencies or lack of capabilities cannot be excluded nowadays. Examples, not applicable to all thermal-hydraulic system codes, are connected with the modelling of:

- the heat transfer between the free liquid surface and the upper gas-steam space,
- the heat transfer between a hotter wall and the cold liquid down-flowing inside a steam-gas filled region.

Those deficiencies are expected to have an importance only in special transient situations.

3.2. The UMAE Method

The UMAE [19], whose flow diagram is given in Fig. 4, is the prototype method for the description of "the propagation of code output errors" approach. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP.

Considering ITF of reference water cooled reactor, and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So, only the accuracy (i.e. the difference between measured and calculated quantities) is extrapolated.

Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change when increasing the dimensions of the facilities (see right loop FG in Fig. 4).

Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of user and nodalizations upon the output uncertainty is minimized in the methodology. However, user and nodalization inadequacies affect the comparison between measured and calculated trends; the error due to this is considered in the extrapolation process and gives a contribution to the overall uncertainty.

The method utilizes a database from similar tests and counterpart tests performed in ITF, that are representative of plant conditions. The quantification of code accuracy (step 'f' in Fig. 4) is carried out by using a procedure based on the Fast Fourier Transform Based Method (FFTBM, [21]) characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy.

Calculations of both ITF experiments and NPP transients are used to attain uncertainty from accuracy. Nodalizations are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations (see left loop FG in Fig. 4). The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling laws considerations (step 'k' in Fig. 4), leads to the Analytical Simulation Model, i.e. a qualified nodalization of the NPP.

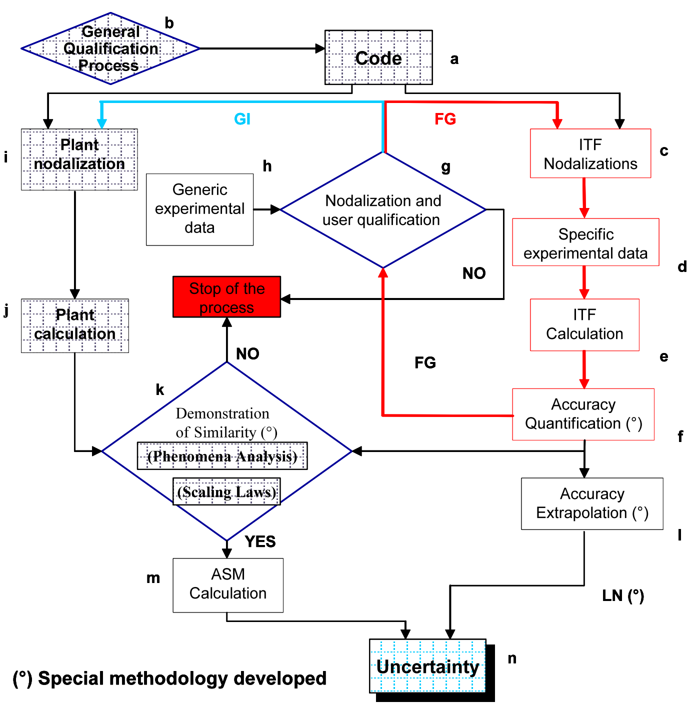

Figure 4   UMAE flow diagram (also adopted within the process of application of CIAU)

3.3. The CIAU Method

All uncertainty evaluation methods are mainly affected by the following limitations:

- The resources needed for their application may be very demanding, ranging up to several man-years;
- The achieved results may be strongly method/user dependent.

The last item should be considered together with the code-user effect, widely studied in the past, and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty method. Therefore, the internal assessment of uncertainty (IAU) was requested as the follow-up of an international conference [10]. The approach CIAU, Code with capability of IAU, has been developed with the objective of reducing the limitations discussed above. CIAU is extensively described in archival technical literature (e.g. ref. [11, 7]) and therefore, only 'spot-information' is given below.

PWR -DRIVING QUANTITIES		(1) Upper Plenum Pressure (MPa)	(2) Primary Circuit Mass Inventory (%)[a]	(3) Steam Generator Pressure (MPa)	(4) Cladding Temperature (K)	(5) Core Power (%)[a]	(6) Steam Generator Level (%)[a]
Hypercube Intervals	1	0.09 – 0.5	10 – 40	0.1 – 3.0	298 – 473	0.5 – 1.0	0 – 50
	2	0.5 – 2.0	40 – 80	3.0 – 7.0	473 – 573	1.0 – 6.0	50 – 100
	3	2.0 – 4.0	80 – 100	7.0 – 9.0	573 – 643	6.0 – 50	100 – 150
	4	4.0 – 5.0	100 – 120	-	643 – 973	50 – 100	-
	5	5.0 – 7.0	-	-	973 – 1473	100 – 130	-
	6	7.0 - 9.0	-	-	-	-	-
	7	9.0 – 10.0	-	-			
	8	10.0 – 15.0	-	-			
	9	15.0 – 18.0	-	-	-	-	-

[a]: Percent of the Initial (nominal) Value

Table 1 CIAU method: Subdivision of driving quantities into intervals

The basic idea of the CIAU can be summarized in two parts:

- Consideration of plant status: each status is characterized by the value of six "driving" quantities (their combination is the "hypercube") and by the time instant when those values are reached during the transient;
- Association of uncertainty (quantity and time) to each plant status.

A key feature of CIAU is the full reference to the experimental data. Accuracy from the comparison between experimental and calculated data is extrapolated to obtain uncertainty. A solution to the issues constituted by the "scaling" and "the qualification" of the computational tools is embedded into the method [6, 4] through the UMAE methodology that constitutes the engine for the development of CIAU and for the creation of the error database.

Assigned a point in the time domain, the accuracy in predicting the time of occurrence of any point is distinguished from the accuracy that characterizes the quantity value at that point.

Thus, the time-domain and the phase-space are distinguished: the time-domain is needed to characterize the system evolution (or the NPP accident scenario) and the phase-space domain is used to identify the hypercubes. The safety relevance and the consistency with the technological achievements have been considered when selecting the driving quantities in Tab. 1. The upper and the lower boundaries have been fixed together with a minimum-optimal number of intervals determined considering: a) design of primary system plant; b) design and licensing of ECCS; c) design and optimization of emergency operational procedures; d) benchmarking of simplified models; e) training purpose; f) code limitations.

Quantity and time accuracies are associated to errors-in-code-models and uncertainties-in-boundary-and-initial-conditions including the time sequence of events and the geometric model of the problem. Thus,

a) The ‘transient-time-dependent’ calculation by a code resembles a succession of steady-state values at each time step and is supported by the consideration that the code is based on a number and a variety of empirical correlations qualified at steady-state with assigned geometric discretization. Therefore, quantity accuracy can be associated primarily with errors-in-code-models.

b) Error associated with the opening of a valve (e.g. time when the equivalent full flow area for the flow passage is attained) or inadequate nodalization induce time errors that cannot be associated to code model deficiencies. Therefore, time accuracy can be associated primarily with uncertainties-in-boundary-and-initial-conditions.

Once the Time Accuracy (Uncertainty) Vector, TAV (TUV), and the Quantity Accuracy (Uncertainty) Matrix, QAM (QUM) are derived, the overall accuracy (and uncertainty) is obtained by the geometric combination of the two accuracies (and uncertainties) values, i.e. time and quantity, in the two-dimensional space-time plane.

4. Selected CIAU applications

Following the CIAU proposal [11], a dozen applications to problems of industrial interest or relevant to the qualification of the method have been completed. Results from the three cases are outlined here.

In the first case the CIAU application was requested by the regulatory authority (or ‘licensor’) in Brazil, [23], within the context of the licensing of the four loop PWR NPP of Angra-2. The results, [24], are given in Fig. 5. Three BEPU (Best Estimate Plus Uncertainty) results are documented, one proposed by the ‘applicant’ (or the ‘licensee’) and two derived by the CIAU. Each result includes one center point (the BE PCT) and upper and lower uncertainty bands. The CIAU results allowed the approval by the licensor of the applicant data: the error bands calculated by CIAU are close (difference less than 20 K) to the error bands calculated by a method (the applicant one) based on the combination of input and output error propagation. The BE PCT proposed by the applicant was not the result of a best estimate calculation, but an average of a number of code runs.

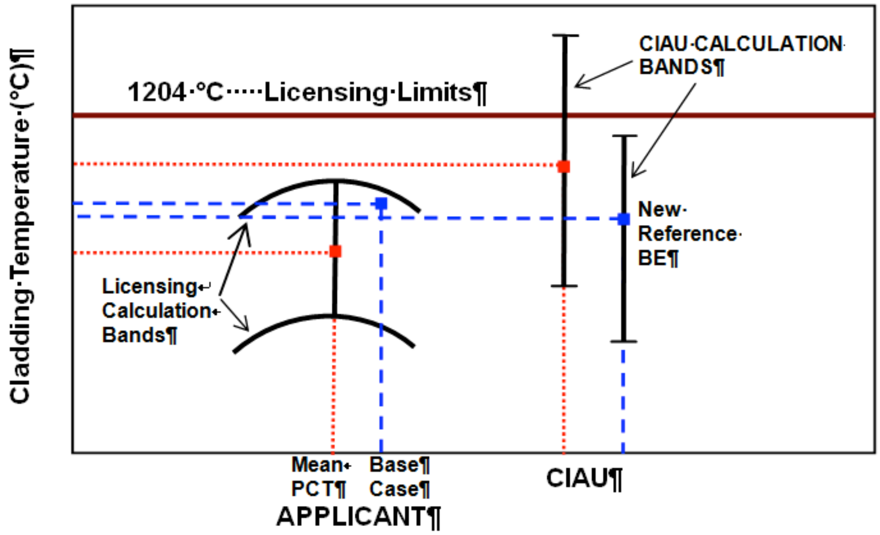

Figure 5 LBLOCA licensing study for Angra-2 PWR

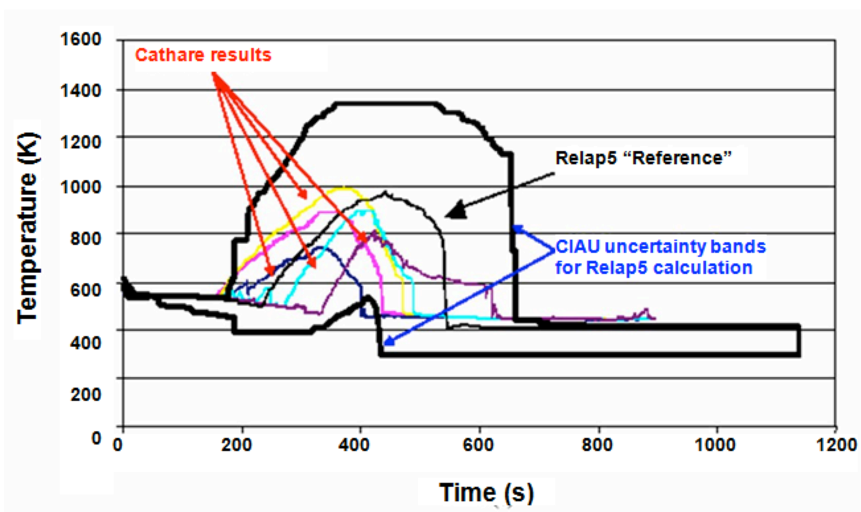

Figure 6 CIAU application to demonstrate that Cathare and Relap5 results coincide as far as the prediction of a safety relevant parameter is concerned

In the second case, [25], the CIAU application was requested by the electrical utility in Bulgaria. The problem consisted in the demonstration that results of two different thermal-hydraulic codes, Cathare and Relap5, coincide as far the computation of a safety relevant parameter was concerned. The reference reactor was the VVER-440 unit 3 of Kozloduy and the concerned transient was a “200 mm break” in cold leg. In order to address the question, Fig. 6, a reference calculation was performed with one of the code (Relap5). Then uncertainty bands were derived by CIAU in relation to the output of the first code and the calculation by the second code (Cathare) was performed. The success of the application consisted in demonstrating that uncertainty bands of one code- calculation (Relap5) envelope the results from the other (Cathare) code-calculation.

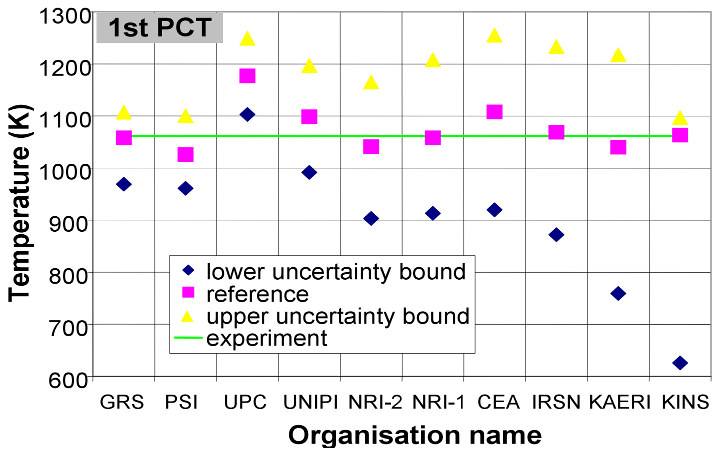

Figure 7 Outcome of the BEMUSE project: uncertainty bounds from each participant ranked by increasing band width from left to right related to the 1st PCT' of the LOFT experiment L2-5

The third selected CIAU application constitutes a qualification study, which at the same time allows a comparison with results of different uncertainty methods. At the international level, within the OECD (Organization for Economic Cooperation and Development) framework, two main activities have been performed (actually the second one is still in progress) as already mentioned: the UMS and the BEMUSE, [8] and [18], respectively. The objective of the project was to predict the LBLOCA performance of the LOFT experimental nuclear reactor (i.e. test L2-5). The process included two steps: the derivation of a reference calculation, involving a detailed comparison between experimental and calculated data, and the derivation of uncertainty bands enveloping the reference calculation. The success of the application consisted in demonstrating that the uncertainty bands envelope the experimental data. Ten international groups participated to the activity [18]. A sample result from the BEMUSE project is outlined in Fig. 7.

The application of the CIAU was performed by the UNIPI while all other participants used an uncertainty method based on the propagation of the input errors supplemented by the use of the Wilks formula. The consistency between the CIAU results and the experimental data can be observed as well as the spread of results obtained by the use of Wilks formula.

Conclusions

The uncertainty evaluation constitutes the ending necessary step for the application of a system thermal-hydraulic code to the nuclear technology. Therefore, any application of a best estimate code without the uncertainty evaluation is meaningless because an error is unavoidable for any prediction. The differences between accuracy, uncertainty and sensitivity have been emphasized and the origins of, or the reasons for, uncertainty (see e.g. ref. [10]) should be clearly in mind when developing an uncertainty approach.

Three main independent ways have been described in the paper to evaluate the uncertainty:

- The propagation of code input errors: this can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code.

- The propagation of code output errors: this is the only demonstrated independent working alternative to the previous one and has also been used for industrial applications. It makes full and direct reference to the experimental data and to the results from the assessment process to derive uncertainty. In this case the uncertainty prediction is not propagated throughout the code.

The deterministic approach based on the ASAP and GASAP extended to performing uncertainty evaluation in conjunction with Data Adjustment and Assimilation: all parameters that affect any prediction, being part of either the code models or the input deck can be considered; proper experimental observations are needed to provide an improved estimate of the probability distribution functions of those parameters through the combination with code predictions and the respective errors. The reduction of the uncertainties in both the system parameters and responses is obtained by the Bayesian inference procedure that is at the basis of Data Adjustment and Assimilation.

The maturity of the methods at the first two bullets may be considered as proved also based upon applications completed within the framework of initiatives of international institutions (OECD/NEA and IAEA). The method at the third bullet constitutes an innovative uncertainty procedure but should not yet be considered as an established technology. However, it constitutes an established idea and framework to pursue a mathematically based road to evaluate the uncertainty in system code predictions.

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