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Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation

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BEST ESTIMATE SAFETY ANALYSIS FOR NUCLEAR POWER PLANTS: UNCERTAINTY EVALUATION

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FOREWORD

Deterministic safety analysis (frequently referred to as accident analysis) is an important tool for confirming the adequacy and efficiency of provisions within the defence in depth concept for the safety of nuclear power plants. Requirements and guidance pertaining to the scope and content of accident analysis have been described in various IAEA publications. To a certain extent, accident analysis is covered in several publications of the revised Safety Standards Series, mainly in the Safety Requirements on design (Safety of Nuclear Power Plants: Design, Safety Standards Series No. NS-R-1) and in the Safety Guide on Safety Assessment and Verification for Nuclear Power Plants (Safety Standards Series No. NS-G-1.2). More detailed guidance has been included in the IAEA safety report on Accident Analysis for Nuclear Power Plants (Safety Reports Series No. 23). The safety report covers all the steps required for accident analyses (i.e. selection of initiating events and acceptance criteria, selection of computer codes and modelling assumptions, preparation of input data and presentation of the calculation results).

The aforementioned safety standards and safety report recommend as one of the options for demonstrating the inclusion of adequate safety margins the use of best estimate computer codes with realistic input data in combination with the evaluation of uncertainties in the calculation results. For the evaluation of uncertainties, the sharing of experience and provision of guidance are elements of vital importance.

This report has therefore been developed to complement the safety standards and the safety report referred to above. It provides more detailed information on the methods available for the evaluation of uncertainties in deterministic safety analysis for nuclear power plants and provides practical guidance in the use of these methods. This report is directed towards analysts coordinating, performing or reviewing best estimate accident analysis for nuclear power plants, both on the utility side and on the regulatory side. It also provides background material for relevant IAEA activities such as seminars, training courses and workshops.

Thanks are due to V. Landauer for the preparation of the manuscript. The IAEA officer responsible for this publication was S. Lee of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Deterministic safety analysis is an essential tool for demonstrating the safety of nuclear power plants. Requirements and guidance pertaining to the scope and content of accident analysis have been described in IAEA Safety Standards Series Nos NS-R-1 [1], NS-R-2 [2] and NS-G-1.2 [3] and in IAEA Safety Reports Series No. 23 [4]. The Safety Guide on safety assessment [3] offers two acceptable options for demonstrating that safety is ensured with an adequate margin, namely the use of best estimate (BE) computer codes combined with conservative input data or combined with realistic input data. Both of these options include evaluation of the uncertainties of results. The second option is particularly attractive because it allows for a more precise specification of safety margins and thus leads to greater operational flexibility.

Prior to having the capability to calculate the uncertainty of key values that define a nuclear power plant's operational envelope, conservative calculations were performed instead. For the present operational plants (mostly Generation II systems that have water as their working fluid), the most important limiting parameter is arguably the peak cladding temperature (PCT), since this parameter defines the threshold at which fuel damage is likely to occur. To the degree that the fuel cladding temperature exceeds the specified limiting value, the probability and extent of core damage, including cladding rupture and fission product release, increases.

The absolute requirement to ensure nuclear power plant core integrity for all events, both abnormal and normal, prompted the regulatory requirement that an acceptable safety margin be formulated and imposed on the operation of nuclear power plants.

In the USA, prior to the existence of Appendix K to Title 10 Part 50 of the Code of Federal Regulations (10 CFR 50) [5], interim acceptance criteria were the vehicle used (up to 1974) to define the plant operational requirements and also the calculation requirements for ensuring that the safety limits were not exceeded. Some of these criteria were plant specific. After that, the regulatory bodies required that all calculations of the limiting parameters, such as the PCT, be performed using specified conservative procedures. An example is the use of conservatively estimated power rating factors to determine the fuel linear power.

In 1974, the first formulation of 10 CFR 50 with sections specifically applicable to nuclear power plant licensing requirements was released. Over a decade later, 10 CFR 50.46 allowed the use of BE codes[1] instead of conservative code models, stipulating, however, that uncertainties be identified and quantified. Other countries established similar 'conservative' procedures and acceptance criteria. Since conservative methods were used to calculate the peak values of key parameters such as the PCT, it was always acknowledged that a large margin existed between the conservatively calculated value and the 'true' value.

BE codes are used in licensing, and conservative initial and boundary conditions are applied without uncertainty quantification. It is claimed that the uncertainties of models are covered by the conservative initial and boundary conditions. Since many utilities apply for power increases, some licensing criteria are approached. One concern is related to the magnitude of model uncertainties and to the determination of the licensing criteria margin reduction.

While the licensing regulations were being codified, an international effort was initiated in parallel to:

(a) Develop BE system analysis codes with the capability to calculate accurate values of the key phenomena that restrict plant operational limits;
(b) Obtain data to enable validation and verification of the system analysis codes to be accomplished;
(c) Perform code validation and verification to ensure that the capabilities of the code are known and acceptable.

The effort to generate relevant data was subdivided into experiments defined to study entire transients (in integral test facilities (ITFs)) and experiments designed to study important phenomena in relative isolation from other phenomena (in separate effects test facilities (SETFs)).

The effort to produce comprehensive data sets for the validation and verification of system analysis codes resulted in the rigorous study and division of plant systems transients into phases that differed by the governing phenomena and dominant plant behavioural characteristics. For example, the earliest phase of a large break loss of coolant accident (LB LOCA) for a Generation II plant is characterized by a rapid depressurization, large loss of primary system inventory, loss of cooling to the core fuel rods and core heat-up. The next phase differs as a function of many phenomena characteristic of

[1] Made possible by the substantial experimental evidence recorded during the intervening years.

emergency core cooling system (ECCS) intervention; inventory begins to accumulate and refill the primary system. Consequently, a clear boundary exists between the two early LB LOCA phases. Using this reasoning process, the entire transient, and in fact all relevant transients, are partitioned into phases that contain ‘phenomenological windows’, which, in turn, leads to the construction of a phenomenologically based system code validation matrix. A number of validation matrices have been developed for various code applications by the Committee on the Safety of Nuclear Installations (CSNI) of the OECD Nuclear Energy Agency (OECD/NEA) [6–8].

With the completion of the CSNI code validation matrices [6] in 1989, the enormous experimental database was categorized according to transient phase and dominant phenomena to both correlate the available data to the code validation needs and to highlight the areas that required further experimental investigation.

With the creation of a database that includes experimental results from a multitude of experiments and the creation of BE system analysis codes such as ATHLET [9], CATHARE [10], RELAP5 [11] and TRAC [12, 13], the components necessary to implement a methodology for calculating the uncertainty of parameters calculated using the BE codes became available.

One of the first uncertainty methodologies presented was code scaling, applicability and uncertainty (CSAU) evaluation [14]. Application of the CSAU methodology resulted in the calculation of the PCT during an LB LOCA design basis accident (DBA) event for a Westinghouse four-loop pressurized water reactor (PWR) with the uncertainty at a 95% confidence level. The PCT was calculated using the TRAC thermohydraulic analysis code and was given as a single valued number with uncertainty bands. The results of this work, first published in 1989, were a defining event for the nuclear safety community. Subsequently a CSAU analysis using RELAP5 was performed for a small break loss of coolant transient on a Babcock & Wilcox PWR.

In the meantime, a number of uncertainty methodologies have been developed in other countries, including the method developed by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) [15], the uncertainty methodology based on accuracy extrapolation (UMAE) method [16] and the AEA Technology (AEAT) method [17]. These methods, although sharing a common goal with CSAU, use different techniques and procedures to obtain the uncertainties on key calculated quantities. More importantly, these methods have progressed far beyond the capabilities of the early CSAU analysis. At present, uncertainty bands (both upper and lower) can be calculated for any desired quantity throughout the transient of interest, in addition to point values like the PCT. One method, namely the internal assessment of uncertainty [18], also includes the capability to assess the calculation uncertainty in a code subroutine while the transient progresses.

The motivation to use BE system analysis codes and to calculate the uncertainty of the final results is compelling. When the calculated values of key parameters such as the PCT are known with great confidence for limiting transients, plant operators may fully exploit a large number of techniques to maximize plant operational efficiencies, power output and plant operational cycles. These capabilities, in turn, enable a utility to reduce the cost of operating a plant.

BE calculations, with quantified uncertainties of key values that describe nuclear power plant behaviour, have been the ultimate goal of nuclear research and development programmes from the start. Only now, after performing research since the 1960s, has the nuclear community achieved this goal. Uncertainty quantification has been and will be used mainly in two different areas, with the following objectives:

(i) To investigate the effect of various input uncertainties on the calculational results obtained with complex thermohydraulic codes;
(ii) To perform uncertainty analyses for licensing purposes.

The means for obtaining the quantified uncertainties for such calculations, employing the most widely used methodologies, are described in this report.

1.2. OBJECTIVES AND SCOPE

There are a number of IAEA publications devoted to the qualified use of advanced computer codes for safety analysis. The objective of this report is to provide practical guidance for the evaluation of uncertainty as a necessary component of BE safety analysis and to encourage the broader use of this approach. Such guidance is considered most important to avoid misinterpretation of calculation results and an unjustified reduction in safety. This report is based on up to date experience worldwide.

In essence, the following items are included: (a) descriptions of and comparisons between the most common methodologies for determining the uncertainties for BE calculations performed to describe nuclear power plant behaviour during transients, accidents and abnormal transients; (b) examples of the use of the methodologies; (c) a summary description of the techniques used to qualify the uncertainty calculations; and (d) a summary of the present trends in this research area. Above all, this report is designed to guide analysts who seek to apply their chosen uncertainty methodology to a system analysis code calculation.

The main use of the BE of safety analysis and, consequently, of this report is expected to be in applications for design and licensing purposes, both for new reactor projects and for periodic safety reviews, safety upgrading and lifetime extensions of existing nuclear power plants.

1.3. STRUCTURE

Uncertainties considered in the IAEA safety standards and other guidance publications are summarized in Section 2. Various options for combining computer codes and input data for safety analysis are also discussed.

Section 3 provides an overview of uncertainty methods (i.e. classification of the methods, propagation of input uncertainties, extrapolation of output error, etc.). Probabilistic and deterministic methods are introduced for the propagation of input uncertainties. Basic features of uncertainty methods that are representative of the classification of the methods are compared.

Steps to ensure adequate uncertainty evaluation methods are discussed in Section 4. Code adequacy, the code application user effect, platform independence and qualification of the methods are considered as major topics. Section 5 suggests various application methods for uncertainty evaluation. This section describes the basic steps in the process, characterization of the scenario, selection of the code, the nodalization process, selection of the uncertainty quantification process and the application of the uncertainty process.

Section 6 deals with the current trends in uncertainty study (i.e. code coupling and internal assessment of uncertainties). The main conclusions and recommendations for the application of uncertainty evaluation in BE safety analysis for nuclear power plants are provided in Section 7.

Five annexes are included: Annex I describes the sources of uncertainty; Annex II summarizes the methods that support the uncertainty methods; Annex III describes the uncertainty methods and gives examples of results; Annex IV discusses the internal assessment of uncertainty at the University of Pisa; and Annex V provides examples of licensing applications.

2. DISCUSSION OF UNCERTAINTIES IN IAEA SAFETY STANDARDS

2.1. CONSIDERATION OF UNCERTAINTIES IN IAEA REFERENCE PUBLICATIONS

Deterministic safety analysis and the consideration of uncertainties, in particular for design and licensing applications, are addressed in IAEA Safety Standards Series Nos NS-R-1 [1] and NS-G-1.2 [3]. Specifically, Ref. [1] requires that: "A safety analysis of the plant design shall be conducted in which methods of both deterministic and probabilistic analysis shall be applied." Further, it is required that: "The computer programs, analytical methods and plant models used in the safety analysis shall be verified and validated, and adequate consideration shall be given to uncertainties."

Moreover, the use of BE codes is generally recommended for deterministic safety analysis in Ref. [3]. Two options are offered to demonstrate sufficient safety margins in using BE codes:

(a) The first option is the use of the codes "in combination with a reasonably conservative selection of input data and a sufficient evaluation of the uncertainties of the results." In this statement, evaluation of uncertainties is meant more in the deterministic sense: code to code comparisons, code to data comparisons and expert judgements in combination with sensitivity studies are considered as typical methods for the estimation of uncertainties.

(b) The second option is the use of the codes with realistic assumptions on initial and boundary conditions. However, for this option "an approach should be based on statistically combined uncertainties for plant conditions and code models to establish, with a specified high probability, that the calculated results do not exceed the acceptance criteria."

Both options should be complemented by sensitivity studies, which include systematic variation of the code input variables and modelling parameters with the aim of identifying the important parameters required for the analysis and "to show that there is no abrupt change in the result of the analysis for a realistic variation of inputs ('cliff edge' effects)."

2.2. SOURCES OF UNCERTAINTY

The requirements and recommendations documented in Refs [1, 3] stem from the presence of uncertainties that have many sources. However, the sources of uncertainty fall within five general categories (see Fig. 1):

(a) Code or model uncertainties: Approximations such as including only some terms in the field equations (e.g. the viscous stress terms are sometimes not included), uncertainties in material properties and the assumption that fully developed flow exists in the system are included in this group of uncertainties.
(b) Representation uncertainties: The discretization of the system (other terms for this include the uncertainty associated with the nodalization or mesh cells representation of the system) to obtain the control volumes that are represented by the field equations.
(c) Scaling uncertainty: Using data recorded in scaled experiments and the reliance on scaling laws to apply the data results to full scale systems.
(d) Plant uncertainty: The uncertainty bands associated with the boundary and initial conditions for the nuclear power plant condition under consideration, for example core power.
(e) User effect: The variation in both the way a number of users will: (i) create and apply a system analysis code and (ii) misapply the system analysis code (i.e. user errors).

A more detailed discussion of the sources of error is given in Annex I.

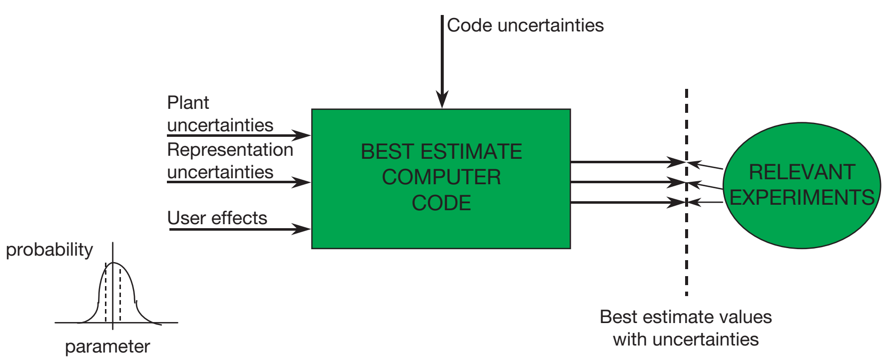

FIG. 1. Evaluation process and main sources of uncertainties.

2.3. SAFETY MARGINS, SENSITIVITY AND UNCERTAINTY

The main objective of safety analysis is to demonstrate in a robust way that all safety requirements are met; that is, that sufficient margins exist between the real values of important parameters and the threshold values at which the barriers against release of radioactivity would fail. The concept of safety margins is presented in Fig. 2.

As shown in Fig. 2, there are two ways to define safety margins (see also the Definitions): either in absolute terms in relation to the expected damage to safety barriers or in relation to acceptance criteria typically set up by the regulatory body. Within the framework of this report, only margins to acceptance criteria will be considered further. Figure 2 also illustrates the difference between the results of conservative and BE analysis. While in a conservative approach the results are expressed in terms of a set of calculated conservative values of parameters limited by acceptance criteria, in a BE approach the results are expressed in terms of uncertainty ranges for the calculated parameters.

Reference [3] recommends performing both sensitivity analysis and uncertainty analysis. It is important to underline that sensitivity analysis must not be misinterpreted as evaluation of the uncertainties (see also Ref. [4]). Sensitivity analysis means evaluation of the effect of variation in input or modelling parameters on code results. Uncertainty analysis means the deviation of quantitative statements on the uncertainty of computer code results from the uncertainties of the input parameters propagated through the model. Another way to derive the uncertainties in code results is by directly

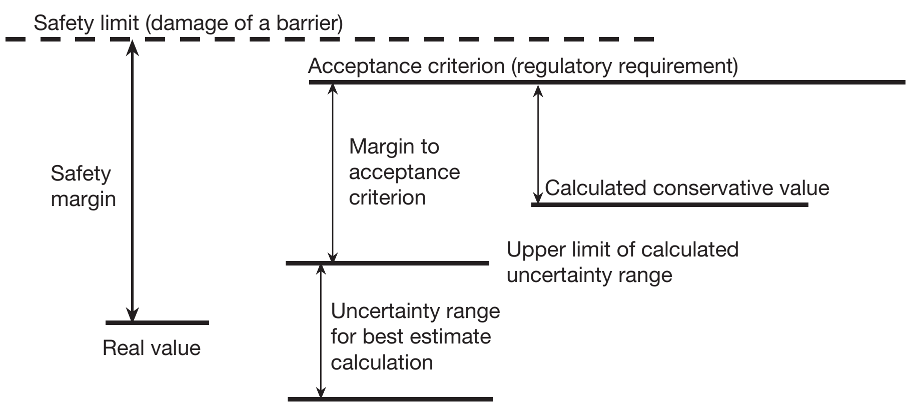

FIG. 2. Concept of safety margins.

comparing code results with experimental data. It is a statistical combination of code uncertainties, representation uncertainties and plant data uncertainties. These two analyses may coincide only under special conditions, when there is very weak interdependence between various uncertain input parameters.

2.4. CONSERVATIVE APPROACH VERSUS UNCERTAINTY EVALUATION

Table 1 summarizes the various options for combining computer codes and input data for safety analysis. Uncertainties are introduced in the calculation both through the computer code and through input data for the code.

A fully conservative approach (option 1) was introduced to cover uncertainties due to the limited capability for modelling physical phenomena based on the level of knowledge in the 1970s. The results obtained by this approach may be misleading (e.g. unrealistic behaviour may be predicted, order of events may be changed). In addition, the level of conservatism (quantified safety margins) is unknown. The use of this approach is therefore no longer recommended in the IAEA safety standards [1].

Options 2 and 3 are considered as acceptable and are suggested in the existing IAEA safety standards [3]. At present, option 2 is still more typically used for safety analysis in many countries. It is reasonably established and its use is straightforward; in some cases just one calculation is sufficient to demonstrate safety. International code validations, as well as various studies on the evaluation of representation and plant data uncertainties, and sensitivity studies help to establish confidence in robustness in the predicted nuclear power plant behaviour. In the USA, however, the Code of Federal Regulations

TABLE 1. VARIOUS OPTIONS FOR COMBINING A COMPUTER CODE AND INPUT DATA

Option	Computer code	Availability of systems	Initial and boundary conditions
1	Conservative	Conservative assumptions	Conservative input data
2	BE	Conservative assumptions	Conservative input data
3	BE	Conservative assumptions	Realistic input data with uncertainties
4	BE	Probabilistic safety analysis based assumptions	Realistic input data with uncertainties

(CFR) does not permit option 2. 10 CFR 50.46 [5] allows either option 3 — to use a BE code plus identification and quantification of uncertainties — or the conservative option 1 (using conservative computer code models). In many cases, a conservative approach is used to avoid the cost of developing a realistic model. However, this approach provides only a rough estimate of the uncertainties; many preparatory calculations are often needed to support a conservative selection of input data, yet an intentionally conservative approach still may not lead to conservative results[2].

Option 4 combines the use of a BE computer code with the specified systems' availability that stems from probabilistic safety analysis assumptions.

This report is aimed at facilitating the wider use of full BE analysis (option 3). BE analysis with evaluation of uncertainties is the only way to quantify the existing safety margins. Its broader use in the future is therefore envisaged, although its application is not always feasible because of the difficulty of quantifying code uncertainties with a sufficiently narrow range for every phenomenon and for each accident sequence.

The current IAEA safety standards [3] allow for the BE selection of both categories of input data with associated evaluation of uncertainties. Thus the availability of nuclear power plant systems could also be judged based on realistic considerations. Even though such considerations are not excluded, in BE analyses performed to date, it is typical to apply evaluation of uncertainties only to physical models embedded in the computer code and to nuclear power plant initial and boundary conditions, while assumptions regarding the availability of nuclear power plant systems are still used in a conservative way. Therefore, a significant conservative component still remains in present BE analyses.

[2] An example is the assumption of high power during a small break LOCA. This overpredicts the swell level in the core and thus leads to better core cooling — the opposite to a conservative requirement. Different sets of conservative assumptions are typically required for each of the acceptance criteria, and different assumptions may even be needed for different time periods of a transient.

3. OVERVIEW OF UNCERTAINTY METHODS

3.1. CLASSIFICATION OF UNCERTAINTY METHODS

An uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). These two main items may be treated differently by different methods. The nine methods[3] discussed herein will illustrate this point.

Within the uncertainty methods considered, uncertainties are evaluated using either (a) propagation of input uncertainties or (b) extrapolation of output uncertainties. For the 'propagation of input uncertainties', uncertainty is obtained following the identification of 'uncertain' input parameters with specified ranges or/and probability distributions of these parameters, and performing calculations varying these parameters. The propagation of input uncertainties can be performed by either deterministic or probabilistic methods.

For the 'extrapolation of output uncertainty' approach, uncertainty is obtained from the (output) uncertainty based on comparison between calculation results and significant experimental data.

These two approaches are illustrated in Fig. 3, and a description of the most commonly used uncertainty methods is provided in the following sections.

3.2. PROPAGATION OF INPUT UNCERTAINTIES

3.2.1. Probabilistic methods

Probabilistic methods include: CSAU, GRS, IPSN, ENUSA, GSUAM and the Canadian best estimate and uncertainty (BEAU) method. The probabilistic methods have the following common features:

(a) The nuclear power plant, the code and the transient to be analysed are identified;
(b) Uncertainties (plant initial and boundary conditions, fuel parameters, modelling) are identified;

[3] AEAW, Canadian BEAU, CSAU, EDF–Framatome, ENUSA, GRS, GSUAM, IPSN and UMAE.

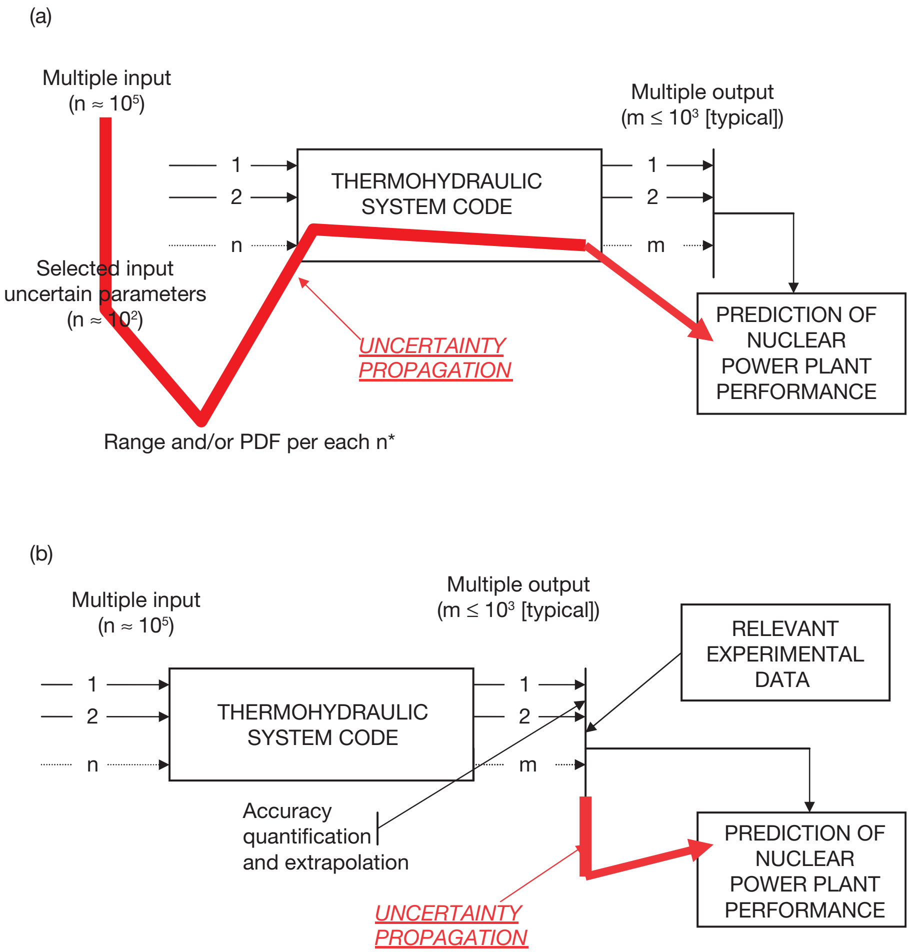

FIG. 3. Uncertainty classification. (a) Propagation of input uncertainties; (b) propagation of output uncertainties.

(c) The methods restrict the number of input uncertainties to be included in the calculations.

The selected input uncertainties are ranged using relevant separate effects data. The state of knowledge of each uncertain input parameter within its range is expressed by a probability distribution. Sometimes ‘state of knowledge uncertainty’ is referred to as ‘subjective uncertainty’ to distinguish it from uncertainty due to stochastic variability. Dependencies between uncertain input parameters should be identified and quantified provided that these dependencies are relevant. Details specific to each probabilistic method are described in the sections that follow.

3.2.1.1. CSAU

The aim of the CSAU methodology is to investigate the uncertainty of safety related output parameters. (In the demonstration cases these were only single valued parameters, such as the PCT or minimum water inventory, with no time dependent values.) Prior to this, a procedure is used to evaluate the code's applicability to a selected plant scenario. Experts identify all the relevant phenomena. Following this step, the most important phenomena are identified and are listed as 'highly ranked' phenomena, based on an examination of experimental data and code predictions of the scenario under investigation. In the resulting phenomena identification and ranking table (PIRT), ranking is accomplished by expert judgement. The PIRT and code documentation are evaluated and it is decided whether the code is applicable to the plant scenario. The CSAU methodology is described in detail by Boyack et al. [14]. Further applications have been performed for an LB LOCA and a small break (SB) LOCA for a PWR [19–21].

All necessary calculations are performed using an optimized nodalization to capture the important physical phenomena. This nodalization represents a compromise between accuracy and cost, based on experience obtained by analysing separate effects tests (SETs) and integral experiments. No particular method or criteria are prescribed to accomplish this task.

Only parameters important for the highly ranked phenomena are selected for consideration as uncertain input parameters. The selection is based on a judgement of their influence on the output parameters. Additional output biases are introduced to consider the uncertainty of other parameters not included in the sensitivity calculations.

Information from the manufacture of nuclear power plant components as well as from experiments and previous calculations was used to define the mean value and probability distribution or standard deviation of uncertain parameters for both the LB and the SB LOCA analyses. Additional biases can be introduced in the output uncertainties.

Uniform and normal distributions were used in the two applications performed to date. Output uncertainty is the result of the propagation of input uncertainties through a number of code calculations. Input parameter uncertainty can be either due to its stochastic nature (i.e. code independent) or due to imprecise knowledge of the parameter values.

No statistical method for uncertainty evaluation has been formally proposed in CSAU. A response surface approach has been used in the applications performed to date. The response surface fits the code predictions obtained for selected parameters, and is used instead of the original computer code. Such an approach then entails the use of a limited number of uncertain parameters in order to reduce the number of code runs and the cost of analysis. However, within the CSAU framework the response surface approach is not prescribed and other methods may be applied.

Scaling is considered by CSAU, identifying several issues based on test facilities and on code assessment. The effect of scale distortions on main processes, the applicability of the existing database to the given nuclear power plant, the scale-up capability of closure relationships and their applicability to the nuclear power plant range are evaluated at a qualitative level. Biases are introduced if the scaling capability is not provided.

3.2.1.2. GRS

The GRS method has some other important features in addition to those mentioned above:

(a) The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined probability distribution of the uncertain parameters, and code calculations are performed by sampled sets of parameters.
(b) The number of code calculations is determined by the requirement to estimate a tolerance and confidence interval for the quantity of interest (such as the PCT). Following a proposal by GRS, Wilks' formula [22, 23] is used to determine the number of calculations required to obtain the uncertainty bands.
(c) Statistical evaluations are performed to determine the sensitivities of input parameter uncertainties on the uncertainties of key results (parameter importance analysis).

This method has no limit to the number of uncertain parameters to be considered in the analysis. The calculated uncertainty has a well established statistical basis. Statistical tools are used for evaluation of the uncertainty and sensitivity at a reasonable number of calculations, as described by Glaeser [15] and Hofer [24].

For the selected plant transient, the method is applied to an integral effects test (IET) simulating the same scenario prior to the plant analysis. If experimental data are not bounded, the set of uncertain input parameters has to be modified.

Experts identify significant uncertainties to be considered in the analysis, including the modelling uncertainties and the related parameters, and identify and quantify dependencies between uncertain parameters. Probability density functions (PDFs) are used to quantify the state of knowledge of uncertain parameters for the specific scenario. In order to differentiate uncertainty due to imprecise knowledge from uncertainty due to stochastic or random variability, the term 'subjective state of knowledge uncertainty' may be used. Uncertainties of code model parameters are obtained based on validation experience.

The scaling effect has to be quantified as a model uncertainty. Additional uncertain model parameters can be included, or PDFs can be modified, accounting for results from SET analysis.

Input parameter values are simultaneously varied by random sampling according to the subjective PDFs and dependencies between them, if relevant. A set of parameters is provided to perform the required number n of code runs. For example, the 95% fractile and the 95% confidence limit of the resulting subjective distribution of the selected output quantities are directly obtained from the n code results, without assuming any specific distribution. No response surface is used.

Sensitivity measures by using regression or correlation techniques from the sets of input parameters and from the corresponding output values allow ranking of the uncertain input parameters in relation to their contribution to output uncertainty. The ranking of parameters is therefore a result of the analysis, not of prior expert judgement. The 95% fractile, 95% confidence limit and sensitivity measures for continuous valued output parameters are provided.

Upper statistical tolerance limits are the upper β confidence for the chosen α fractile. The fractile indicates the probability content of the probability distributions of the code results (e.g. $\alpha = 95\%$ means that the PCT is below the tolerance limit with at least $\alpha = 95\%$ probability). One can be β% confident that at least α% of the combined influence of all the characterized uncertainties are below the tolerance limit. The confidence level is specified because the probability is not analytically determined. It accounts for the possible influence of the sampling error due to the fact that the statements are obtained from a random sample of limited size. The smallest number n of code runs to be performed is given by Wilks' formula [22, 23]:

$$\left(1-\alpha/100\right)^{n} \geq \beta/100$$

which is the size of a random sample (a number of calculations) such that the maximum calculated value in the sample is an upper statistical tolerance limit. The required number n of code runs for the upper 95% fractile is: 59 at the 95% confidence level, 45 at the 90% confidence level and 32 at the 80% confidence level.

For two-sided statistical tolerance intervals (investigating the output parameter distribution within an interval) the formula is:

$$1-\alpha^{n}-n(1-\alpha)\alpha^{n-1}\geq\beta$$

The minimum number of calculations can be found in Table 2.

For regulatory purposes, where the margin to licensing criteria is of primary interest, the one-sided tolerance limit may be applied; that is, for a 95th/95th percentile, 59 calculations would be performed.

As a consequence, the number n of code runs is independent of the number of selected input uncertain parameters, only depending on the percentage of the fractile and on the desired confidence level percentage. The number of code runs for obtaining sensitivity measures is also independent of the number of parameters. As an example, 100 runs were carried out in the analysis of a reference reactor, using 50 parameters.

3.2.1.3. IPSN

The method developed by the Institut de protection et de sûreté nucléaire (IPSN), France, is basically the same as the GRS method. In the OECD/NEA–CSNI uncertainty methods study (UMS) [17] only 'basic uncertainties' stemming from the constitutive equations in the code were considered. Therefore, all the information provided hereafter in relation to the GRS method also applies to the IPSN method.

TABLE 2. MINIMUM NUMBER OF CALCULATIONS n FOR ONE-SIDED AND TWO-SIDED STATISTICAL TOLERANCE LIMITS

β/α	One-sided statistical tolerance limit			Two-sided statistical tolerance limit		
	0.90	0.95	0.99	0.90	0.95	0.99
0.90	22	45	230	38	77	388
0.95	29	59	299	46	93	473
0.99	44	90	459	64	130	662

3.2.1.4. ENUSA

The method developed by Empresa Nacional del Uranio, SA (ENUSA), Spain, is basically the same as the GRS method and the CSAU framework. Wilks' formula is used, as in the GRS method, and no use of response surfaces has been made. The number of input parameters, however, has been limited to 26 in the UMS application [17] by going through a PIRT process. The reason was to limit the effort required to determine input uncertainty distributions. Therefore, all the information reported in relation to the GRS method applies to the ENUSA method.

3.2.1.5. GSUAM uncertainty method used by Siemens (now Framatome ANP)

The generic statistical uncertainty analysis methodology (GSUAM) constitutes a proprietary uncertainty method developed by Siemens (Framatome ANP). The method was used to support the licensing process of the Angra 2 nuclear power plant [25].

GSUAM aims at the evaluation of point values such as the PCT for the uncertainty, not for time dependent quantification of the uncertainty of code results. The method includes general features similar to the CSAU framework. Three main contributions to uncertainty are identified:

(a) The code;
(b) Nuclear power plant conditions;
(c) Fuel conditions.

Of these elements, the code constitutes the largest source of overall uncertainty. This is obtained from the comparison between experimental and calculated data following an approach similar to the UMAE.

In order to address the remaining uncertainty sources, sensitivity studies are performed following the identification of uncertainty input parameters and the related range of variation. A statistical method is used to combine the uncertainty data obtained from the three uncertainty sources.

3.2.1.6. BEAU method used in Canada

The BEAU method has been developed and applied in Canada by Ontario Power Generation [26] and Atomic Energy of Canada Ltd [27]. The Canadian Nuclear Safety Commission published lessons learned from trial applications and features expected of a BE analysis by regulators [28]. Further applications have been performed to investigate the first power pulse during an

LB LOCA in a CANDU reactor (which has a positive void coefficient). The approach taken is consistent with the CSAU framework and approximately similar to CSAU demonstration applications. A PIRT process is performed, and a response surface is used based on computer code calculations. A large number of calculations were performed using the response surface to replace the computer code. A probabilistic uncertainty statement (i.e. 95th percentile values) is obtained. The main focus is on plant parameter uncertainties. These applications were reviewed by an international expert panel [29].

3.2.2. Deterministic methods

The deterministic methods include the Atomic Energy Authority Winfrith (AEAW) and the Electricité de France (EDF)–Framatome methods. The deterministic methods have the following features in common with probabilistic methods:

(a) The code, nuclear power plant and transient are identified;
(b) Uncertainties (initial and boundary conditions, modelling, plant, fuel) are identified.

The difference with deterministic methods is in quantifying the input parameter uncertainties. No probability distributions are used; instead, reasonable uncertainty ranges or bounding values are specified that encompass, for example, available relevant experimental data. The statements of the uncertainty of code results are deterministic, not probabilistic.

3.2.2.1. AEAW

The AEAW method considers the deterministic nature of most of the processes involved and does not use statistical procedures [14]. For the investigated scenario, experts identify the relevant phenomena and select the most important uncertain parameters. Physical reasons are provided for each selected parameter (i.e. why it could contribute to the uncertainty of the key output parameters).

A reasonable uncertainty range is specified for each parameter, defined as the smallest range of values that includes all the values for which there is no reasonable certainty that they are inconsistent with available evidence.

Experimental data examination supports the characterization phase of modelling uncertainties, generally from SETFs. Bounding models are built in such a way as to predict, for any parameter combination, acceptable upper and lower limits for the assessed quantity. Alternatively, deviations of code predictions compared with SET data are combined, choosing bounding deviations to be included in the code predictions, thus ensuring that all available deviations are bounded.

No general method is proposed to evaluate the range of output uncertainties. Standard and bounding values are used to address the uncertainties. Code runs with single or multiple parameter variations are carried out in order to define those combined variations believed to maximize or minimize the output quantity addressed, thus obtaining reasonable bounding uncertainty ranges. This means that the number of code runs increases with the number of uncertain parameters. During the variation analysis phase, assigning two values for each parameter other than the standard value results in about $2N + 1$ code runs in the case of N parameters. The aim of additional runs is to maximize or minimize the output quantity.

The code applicability to a nuclear power plant calculation is anticipated by using the method for an integral test taken from an independent database to check whether experimental data are within the determined ranges. If they are not within the determined ranges, it is concluded that changes of the input uncertainty ranges or the combination of uncertainties or further code development is necessary. The processes involving scaling effects, modelling and quantification of the related uncertainty are taken into account by expert judgement. The adopted system code must calculate the scale of the various experiments.

3.2.2.2. Method used by EDF–Framatome

EDF and Framatome have developed an accident analysis method [27] based on the use of realistic computer codes, namely the deterministic realistic method (DRM). Its principle is based on quantification of the calculational uncertainty, which is taken into account deterministically when the results (uncertainty parameters) are compared with the acceptance criteria. To ensure that the value of an uncertainty parameter is conservative, a penalization mode is introduced into the realistic model. The penalties are chosen so as to preserve a realistic response from the code. The DRM was applied to an LB LOCA for a French three-loop PWR.

Since publication of the original 10 CFR 50.46 rule in 1974, significant improvements have been made in the understanding and modelling of LOCA phenomena, and the methods specified in Appendix K to demonstrate the acceptability of the ECCS have proved to be overly conservative. Since the revision of 10 CFR 50.46 [5] in 1988, emergency core cooling (ECC) analyses may be carried out with realistic models, provided that the uncertainty in the calculation results is estimated with a high confidence level, ensuring a high probability that the safety criteria will not be reached. In order to cope with the evolution of the rule, EDF and Framatome decided to jointly develop a new methodology, the deterministic realistic methodology, used in association with CATHARE, the French BE code dedicated to thermohydraulic safety analyses [30]. This methodology is based on statistical and deterministic approaches. A statistical analysis quantifies the uncertainties. These uncertainties should be bounded by a deterministic calculation. Through this procedure, the realistic nature of the simulation should be preserved, as shown in Ref. [31]. The DRM is a general approach applicable to all types of accident scenario.

3.2.2.2.1. DRM principles

Two main factors contribute to the code uncertainties:

(a) Uncertainty in the initial and boundary conditions;
(b) Uncertainty in the physical code models.

The objective of the DRM is to quantify the overall uncertainty by means of a statistical analysis. The CATHARE V1.3L code is used as it can provide a BE evaluation of all the most important, dominant physical phenomena of the transient. The resulting realistic plant model is qualified by comparison with relevant experimental tests.

The realistic plant model calculates each output parameter (e.g. the PCT, oxide layer thickness) both at the BE or most probable level and at the 95% probability level. For the 95% probability level, uncertainties of the code and the plant and fuel parameter uncertainties are accounted for.

In the deterministic evaluation model, the uncertainty of the output parameter is bounded by defining a penalization mode that ensures conservative results. The value of the parameter resulting from the DRM approach is therefore higher than the 95% confidence level value of the same parameter calculated using the statistical method.

The pertinence of the penalties introduced into the DRM plant model results from physical and statistical analyses. As far as possible, the penalties are directly assigned to the parameters that generate them, in order to minimize the conservatism and to preserve the realistic response of the code. In this way, the DRM model differs from the previous deterministic Appendix K evaluation models [31], in which the penalization mode was defined a priori. Nevertheless it must be noted that the safety demonstration relies on the 95% confidence level value of the uncertainty parameter. The objective of the deterministic model is only to provide an industrial tool for all application calculations needed for nuclear power plant safety assessment. Implementation of the DRM approach can be divided into four action phases as follows:

(i) Justification of the realistic nature of the model used. The capability of the code to simulate the dominant physical phenomena of the transient is checked. This analysis is based on the code characteristics and assessment (described in the code documentation). The capacity of the reactor model to enable realistic predictions can also be evaluated on the basis of simulating relevant experiments. This analysis can lead to implementation of additional models in the code.
(ii) Estimation of the overall uncertainty. A methodology is applied to quantify the overall uncertainty of the transient uncertainty parameters resulting from combination of the basic uncertainties. The method is derived from the CSAU procedure developed by the US Nuclear Regulatory Commission (NRC) [32]. It focuses on the impact of the dominant phenomena relative to the scenario considered. The basic uncertainties are estimated for the key code models based on comparison between calculations and experiments. The propagation of the basic uncertainties through a reactor calculation is assessed by means of a statistical method. The PDF of the uncertainty parameter is determined using a response surface associated with Monte Carlo random sampling of the elementary parameters. The impact of the biases that are not rectifiable is added to the PDF of the uncertainty parameter, from which the 95% confidence level value is determined. For the LB LOCA, this analysis is carried out for each PCT independently.
(iii) Penalization. The chosen penalization enables the determined uncertainties to be enveloped in a reasonably conservative manner. This is specific to each type of transient and each criterion to be verified. Moreover, the chosen penalization must not distort the prediction of the system effects, which function as boundary conditions for the hot fuel assembly calculation. It is therefore better to introduce the conservatism on the parameters that make the major contribution to the uncertainty but that do not present a risk of altering the system behaviour. For an LB LOCA, the penalization mode is the same for all the PCTs.
(iv) Evaluating the conservatism. Demonstration of the conservative nature of the DRM model, as in the interim approach proposed by the NRC [32], relies on a comparison of the DRM uncertainty parameter values with the 95% confidence level values determined by the statistical analysis. The range of applicability of the DRM model is defined by the list of dominant phenomena considered for the analysis. A verification — or even a new evaluation — of the uncertainty at the 95% confidence level is

required only if the differences in the reactor design characteristics, the characteristics related to nuclear parameters or the technical specifications reveal new dominant physical phenomena (or if they modify the sensitive factors considered in the statistical analysis).

3.2.2.2.2. Penalization mode

The statistical method applied to a realistic plant model enables estimation of the peak temperatures of the fuel rod cladding during an LB LOCA transient with a confidence level greater than 95%. The DRM uses a realistic plant model and introduces a penalization mode that covers the overall calculation uncertainty. The mainspring of the DRM model is to provide a simple industrial tool. It is used to perform application calculations needed for nuclear power plant safety assessment, instead of the statistical method, which remains the reference tool. The principles determining the choice of penalization mode are linked to the objective of non-distortion of the transient physics and that of introducing penalties as close as possible to the sources of uncertainty.

3.3. EXTRAPOLATION OF OUTPUT UNCERTAINTY

3.3.1. Uncertainty methodology based on accuracy extrapolation

The UMAE method focuses not on the evaluation of individual parameter uncertainties but on direct scaling of data from an available database, calculating the final uncertainty by extrapolating the accuracy evaluated from relevant integral experiments to full scale nuclear power plants, as described in Ref. [33].

Considering ITFs of a reference light water reactor (LWR) 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 the reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. Only the accuracy (i.e. the difference between measured and calculated quantities) is therefore 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; however, available IET facility scales are far from reactor scale.

Other basic assumptions are that phenomena and transient scenarios in larger scale facilities are close enough to plant conditions. The influence of the user and the nodalization 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 contributes to the overall uncertainty.

The method uses a database from similar tests and counterpart tests performed in ITFs that are representative of plant conditions. The quantification of code accuracy is carried out by using a procedure based on fast Fourier transform (FFT), characterizing the discrepancies between code calculations and experimental data in the frequency domain and defining figures of merit for the accuracy of each calculation[4]. Different requirements have to be fulfilled in order to extrapolate the accuracy.

Calculations of both IETs and plant transients are used to obtain uncertainty from accuracy. Discretized models and nodalizations are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy be satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations. The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, taking scaling laws into consideration, leads to the analytical simulation model (ASM); that is, a qualified nodalization of the plant.

In applying the method, no limitation is placed on the number of input uncertain parameters. The related input parameter variation ranges are reflected in the output parameter variation ranges; it is not possible to establish a correspondence between each input and each output parameter without performing additional specific calculations, which, however, are beyond the scope of the UMAE. The process starts with the experimental and calculated database. Following the identification (e.g. from the CSNI validation matrix) of the physical phenomena involved in the selected transient scenario, relevant thermohydraulic aspects (RTAs) are used to evaluate the acceptability of code calculations, the similarity among experimental data and the similarity between

[4] FFT, incorporated into the fast Fourier transform based method (see Section III–3 of Annex III), is used for the acceptability check of the calculations. (In the UMAE the ratio of the experimental to calculated value is used for the extrapolation.) The FFT based method is then independent with respect to the philosophy of UMAE; it is used as a tool. The use of this procedure avoids the influence of engineering judgement in evaluating the adequacy of the code results.

plant calculation results and available data. Statistical treatment is pursued in order to process accuracy values calculated for the various test facilities and to obtain uncertainty ranges with a 95% confidence level. These are superimposed as uncertainty bands bracketing the ASM calculation.

The scaling of both experimental and calculated data is explicitly assessed within the framework of the analysis. In fact, the demonstration of phenomena scalability is necessary for the application of the method and for the evaluation of the uncertainty associated with the prediction of the nuclear power plant scenario.

Comparison of thermohydraulic data from experimental facilities of a different scale constitutes the basis of the UMAE. Special steps and procedures are included in the UMAE to check whether the nodalization and code calculation results are acceptable. An adequate experimental database including the same phenomena as in the selected test scenario of the nuclear power plant is needed for the application of this method. For a successful application it is necessary that the accuracy of the calculations does not dramatically decrease with increasing scale of the experimental facilities. The demonstration that accuracy increases when the dimensions of the facility in question are increased (for which a sufficiently large database is required, which is not fully available now) would be a demonstration of the consistency of the basic idea of the method.

Tables of parameters are used to compare and characterize the above uncertainty methods (see Sections 3.4 and 5.5).

3.3.2. Availability of a method for the internal assessment of uncertainty

All of the uncertainty evaluation methods are affected by two main limitations:

(a) Their application may be very resource intensive, demanding up to several person-years;
(b) The results achieved may be strongly method and/or user dependent.

The user dependence of the uncertainty evaluation combined with user effects in the application of thermohydraulic system codes — an issue that has been extensively studied in the past [34] — may undermine the usefulness of uncertainty evaluations. This problem became evident during the International Workshop on Transient Thermal-hydraulic and Neutronic Codes Requirements held in 1996 [35], resulting in the call for a method — referred to as the internal assessment of uncertainty — that would be inherent to a system code.

In response to this need, the University of Pisa has developed a code with the capability of internal assessment of uncertainty (CIAU) [18].

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

(i) Consideration of plant state: each state is characterized by the value of six relevant quantities (i.e. a hypercube) and by the value of the time since the transient start.
(ii) Association of an uncertainty to each plant state.

In the case of a PWR the six quantities are: (a) the upper plenum pressure; (b) the primary loop mass inventory (including the pressurizer); (c) the steam generator pressure; (d) the cladding surface temperature at 2/3 of core active height (measured from the bottom of the active fuel), where the maximum cladding temperature occurring in one horizontal core cross-section is expected; (e) the core power; and (f) the steam generator downcomer collapsed liquid level. If levels are different in the various steam generators, the largest value is considered.

A hypercube and a time interval characterize a unique plant state for the purpose of uncertainty evaluation. All plant states are characterized by a matrix of hypercubes and by a vector of time intervals. Let us define Y as a generic thermohydraulic code output plotted versus time. Each point of the curve is affected by a quantity uncertainty and by a time uncertainty. Owing to the uncertainty, each point may take any value within the rectangle identified by the quantity and time uncertainties. The value of uncertainty — corresponding to each edge of the rectangle — can be defined in probabilistic terms. This satisfies the requirement of a 95% probability level acceptable to NRC staff for comparing BE predictions of postulated transients with the licensing limits in 10 CFR 50.

The idea at the basis of the CIAU may be described more specifically as follows:

- The uncertainty in code prediction is the same for each plant state. A quantity uncertainty matrix (QUM) and a time uncertainty vector (TUV) can be set up including values of U_q and U_t obtained by means of an uncertainty methodology.

Additional information can be found in Annex IV.

3.4. COMPARISON OF BASIC FEATURES OF THE METHODS

In order to identify and characterize relevant input uncertain parameters (e.g. in modelling, boundary or initial conditions), a selection is usually necessary to identify the most important ones. Two of the uncertainty methods considered (CSAU and GRS) define ranges and probability distributions (not required for the AEAW method and the UMAE) for these uncertain parameters. This process is essentially based upon validation of the code models with data from IET and SET facilities. Probability distributions are used to express the state of knowledge about uncertain parameters. PDFs can be combined, accounting for any identified interdependence of the parameters, thus providing the joint PDF.

With reference to the second item (i.e. the quantification of the global influence, obtained from the combination of the input uncertainties, on selected output parameters, for example the PCT), various mathematical procedures are implemented in the different methods.

Depending on the method, the number of code runs necessary for an uncertainty analysis can increase with the number of uncertain parameters considered. This is an index for the evaluation of the computational resources required by the entire analysis and strongly varies from one method to another. The number of uncertain parameters is therefore limited, and only the most important ones are varied (for the AEAW and the CSAU methods, but not for the GRS method); expert judgement and/or simplified analytical methods normally contribute towards making this selection. A systematic investigation of input parameter uncertainties is not the objective of the UMAE.

An important item to be addressed in an uncertainty analysis for a nuclear reactor is the scaling effects and their treatment by the different methods. Code validation is primarily performed by reduced scale data; no evidence is provided regarding their applicability at the scale of a real plant. As far as possible, scaling capability of a code should be demonstrated during the validation process and should be addressed using the uncertainty methods. Scale-up capabilities of closure relations can be proved at least to a partial degree by calculating separate effect experiments performed in the full scale Upper Plenum Test Facility (UPTF) [36] for a spectrum of two phase flow phenomena.

Table 3 presents a comparison of the basic features of the uncertainty methods that characterize the classification of the methods described in the previous sections.

The pioneering development and application of an uncertainty evaluation of computer code results was proposed and performed by the NRC with the CSAU methodology [14]. The framework of CSAU makes it possible to

TABLE 3. COMPARISON OF RELEVANT FEATURES OF UNCERTAINTY METHODS

	Feature	AEAW	CSAU	GRS	UMAE
1	Determination of uncertain input parameters and of input uncertainty ranges	Experts	Experts	Experts	[a]
2	Selection of uncertain parameter values within the determined range for code calculations	Experts	Experts	Random selection	Not necessary
3	Support of identification and ranking of main parameter and modelling uncertainties	No	Yes	No	No
4	Accounting 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 rigour	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 of number of input and output parameters	No	No	Yes	Yes
9	Typical number of code runs	LOBI[b]: 22 LSTF[b]: 50	LB: 8 SB: 34	59 LSTF PWR LOFT[b]:100	n.a.[c]
10	Number of uncertain input parameters	LOBI: 7 LSTF: 9	LB: 7 (+5) SB: 8	LSTF LOFT PWR ≈50	n.a.
11	Quantitative information about influence of a limited number of code runs	No	No	Yes	No
12	Use of response surface to approximate result	No	Yes	No	No

TABLE 3. COMPARISON OF RELEVANT FEATURES OF UNCERTAINTY METHODS (cont.)

	Feature	AEAW	CSAU	GRS	UMAE
13	Use of biases on results	No	Yes	No	For non-model uncertainties
14	Continuous valued output parameters	Yes	No	Yes	Yes
15	Sensitivity measures of input parameters on output parameters	No	No	Yes	No

[a] The differences between experimental and used input data constitute one of the sources of uncertainty.

[b] LOBI, LOFT and LSTF are test facilities.

[c] This depends on the stage of the analysis. The first application to the analysis of the SB LOCA counterpart test in a PWR required roughly 20 code runs; the analysis of a similar nuclear power plant scenario would require a few additional code runs.

n.a.: not applicable.

proceed through the different elements and steps in the process of evaluating uncertainty. This includes requirements and code capabilities, going through a PIRT process to identify the most important phenomena, investigating the suitability of a code to calculate the scenario under investigation, assessment and ranging of parameters, and performing sensitivity and uncertainty analyses. The procedure has been applied in demonstration cases [14, 21]. The last element, uncertainty evaluation, was carried out using a specific method, including probability distributions for uncertain input parameters, response surfaces, etc. This specific application is not required by CSAU, and other methods, such as the use of Wilks' formula, could be applied within the CSAU framework. Consequently, it is not uncommon for uncertainty methodologies to contain some or all of the CSAU framework elements while differing significantly from the procedures used in the original CSAU demonstration and application. In Table 3, for example, the method used for uncertainty evaluation applied in the CSAU application cases is compared with other methods to quantify the uncertainty.

4. QUALIFICATION OF EVALUATION METHODS

Adequate uncertainty evaluation methods must include steps to ensure that: (a) the system code is adequate; (b) the user effect is properly accounted for; (c) the influence of the computational platform is minimized; and (d) the uncertainty methodology is qualified. These topics are described below.

4.1. CODE ADEQUACY

Whether a code is adequate for performing BE plus uncertainty analyses is generally determined by using both top-down and bottom-up evaluations, as summarized in Fig. 4 and outlined here.

4.1.1. Bottom-up code adequacy

Bottom-up evaluation of code adequacy consists of four parts: examination of the pedigree, applicability, fidelity and scalability of the code under consideration.

4.1.1.1. Pedigree

The pedigree of a system code consists of knowing its history, the procedures involved in its development and the basis for each correlation that is used in the code. The correlations used in the code must be documented, for example in textbooks, laboratory reports and papers. The uncertainty data used to bound the correlations, for example instrumentation uncertainty and data system uncertainties, must be included in the documentation. The basis for the uncertainties should be traceable and reproducible. The assumptions and limitations of the models must be known and documented.

4.1.1.2. Applicability

The applicability of a system code includes knowing the range of use for each of its correlations and having these correlations in documented form. The correlations used in the code should be referenced. Finally, the range of applicability claimed in the code manual should be consistent with the pedigree, or, if a greater range is claimed, the justification for the increase in range must be reported.

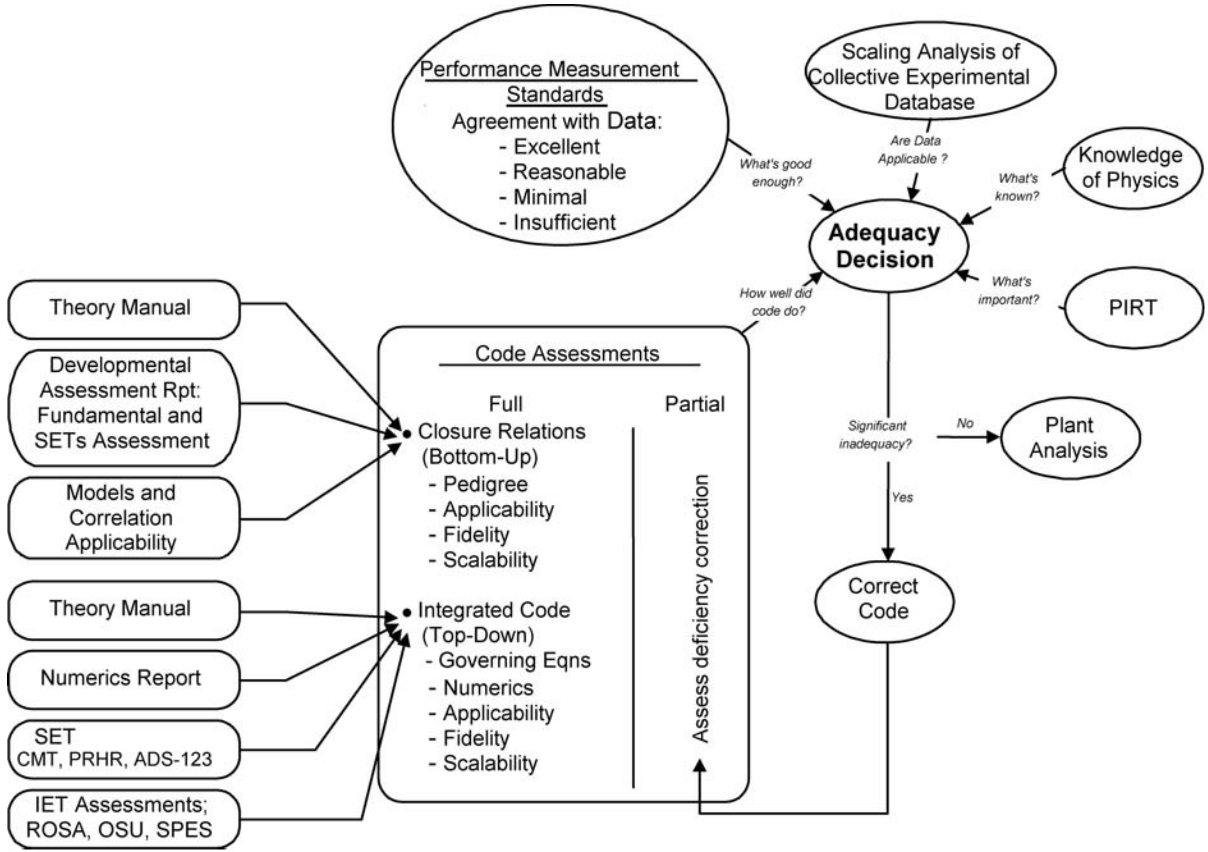

FIG. 4. Process for assuring code adequacy.

4.1.1.3. *Fidelity*

The fidelity of a system code ensures that the correlations used in the code are not altered in an ad hoc manner with respect to their documented formulation. Furthermore, a validation effort designed to measure the code calculation of key phenomena versus data must be performed, and the validation effort should be complete for all key phenomena for the transients of interest. Finally, benchmarking studies may supplement the validation effort if appropriate standards are available, for example comparison of a code calculation with a closed form solution.

4.1.1.4. *Scalability*

Bottom-up scaling stems from the need to:

(a) Build experimental facilities that model the desired full scale system;
(b) Closely match the expected behaviour of the most important transient phenomena in the scenarios of interest;
(c) Demonstrate the applicability of data from a scaled facility to a full scale system;
(d) Defend the use of data from a scaled facility in a code used to calculate the behaviour of a full scale system;
(e) Relate a calculation of a scaled facility to a calculation of a full scale system.

Usually, scalability studies are performed to scale key parameters for a portion of the system behaviour and not to correlate the global system behaviour. Therefore, scalability analyses consist of four steps: (i) isolate the first order phenomena; (ii) characterize the first order phenomena; (iii) convert the defining equations into non-dimensional form; and (iv) adjust the experimental facility conditions to give equivalent (or near equivalent, that is based on non-dimensional numbers that follow from step (iii)) behaviour with the full scale system within the limitation of the facility.

As may be noted from the above discussion, bottom-up code adequacy techniques focus principally on closure relationships.

Thus the field equations used in the code must be correctly formulated and programmed. In addition, the field equations must be reviewed by the scientific community, whose agreement on the correct formulation and insertion of the governing equations in the code must be obtained.

4.1.2. Top-down code adequacy

The top-down approach for ensuring code adequacy focuses on the capabilities and performance of the integrated code. The top-down approach consists of four parts: numerics, fidelity, applicability and scalability.

4.1.2.1. Numerics

The numeric solution evaluation considers: (a) convergence, (b) stability and (c) property conservation[5]. Again, agreement by the scientific community on acceptable convergence, stability and property conservation must be obtained.

4.1.2.2. Fidelity

The fidelity of the code is demonstrated by performing thorough code assessments based on applicable integral effects and separate effects data. The data are part of an agreed upon code assessment matrix constructed based on the transients of importance and the key phenomena for each phase of the transients.

4.1.2.3. Applicability

The code must be shown capable of modelling the key phenomena in the system components and subsystems by conducting thorough validation studies. The key phenomena are identified in the PIRT.

The method to determine whether the code is capable of modelling key phenomena consists of comparing the calculation produced by the code to data that have known uncertainties. For example, excellent agreement between the code calculation and data is shown in Fig. 5, where the calculated value is at all times within the data uncertainty band.

Reasonable agreement between the calculation and data is shown in Fig. 6. The distinction between excellent and reasonable agreement depends on whether the calculated value lies within the experimental uncertainty band. If

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[5] Property conservation issues arise when two calculations of the same property are performed by a system code using two different algorithms or methods. This may result in the need to enhance the accuracy of the code result. Since the two methods are likely to calculate slightly different values of the same property (e.g. pressure), property conservation must be considered.

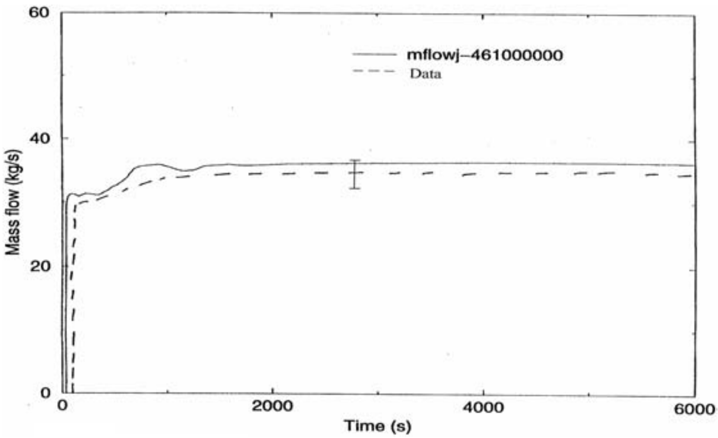

FIG. 5. Example of excellent agreement between code and data.

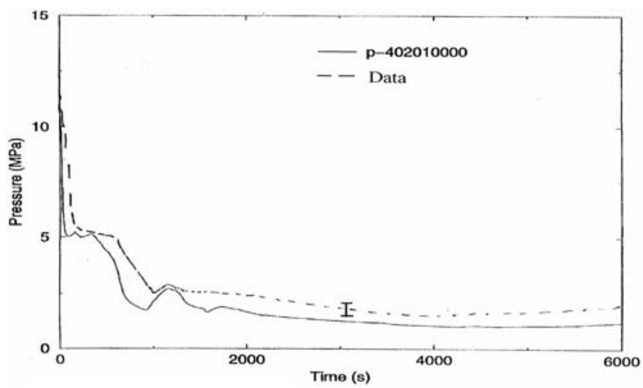

FIG. 6. Example of reasonable agreement between code and data.

the calculated value is outside the experimental uncertainty band, but the calculated value shows fundamentally the same behaviour as the data, then the calculated result is considered reasonable.

The degree of agreement between the code calculation and the data is generally divided into four categories, as shown in Table 4. A code is considered to be of adequate applicability when it shows either excellent or reasonable agreement with the highly ranked phenomena (sometimes identified as the dominant phenomena) for a transient of interest. If the code

TABLE 4. CODE ADEQUACY IDENTIFIERS

Classifier	Description
Excellent	The calculation lies within or near the data uncertainty band at all times during the phase of interest.
Reasonable	The calculation sometimes lies within the data uncertainty band and shows the same trends as the data. Code deficiencies are minor.
Minimal	Significant code deficiencies exist. Some major trends and phenomena are not predicted. Incorrect conclusions may be drawn based on the calculation when data are not available.
Unacceptable	A comparison is unacceptable when a significant difference between the calculation and the data is present — and the difference is not understood. Such a difference could follow from errors in either the calculation or the portrayal of the data — or an inadequate code model of the phenomenon.

gives minimal or unacceptable agreement, additional work must be performed; the work may range from additional code development to additional analysis to understand the phenomena.

4.1.2.4. Scalability

Experimental scaling distortions, for example inappropriate environmental heat losses that stem from the larger surface to volume ratios that are inherent to scaled facilities, are identified and isolated. Finally, an effort to isolate all code scaling distortions is performed through the code assessment calculations. Scaling distortions may arise from inappropriate use of a correlation developed in a small scale system when applied to a full scale system.

4.2. CODE APPLICATION USER EFFECT

The training, level of expertise, calculational objectives and professionalism of the system code user exert a large influence on the calculational results.

4.2.1. Training and level of expertise

All of the system codes share an inherent complexity that follows from the complexity of the problems that are being analysed. For example, the LOCA scenarios that must be analysed encompass a multitude of flow regimes, heat transfer regimes and complex interactions that, unless familiar to the user, are a bewildering array of variables that encourage error, misapplication and misinterpretation. Consequently, training of the user is necessary for each system code, particularly when the user is required to perform licensing calculations. Some kind of user qualification is, in fact, advisable. Following training, the user will normally increase in expertise as a function of the level of effort and the user's fundamental understanding of the physics.

A measure of the importance of the user effect is reflected in a number of international standard problems analyses in which the results obtained by a number of users applying the same system code to the same problem with the same set of initial and boundary conditions are documented.

4.2.2. Calculational objective

Occasionally, different users will produce models of the same system that differ considerably from one another depending on their calculational objective. If a user attempts to perform an order of magnitude calculation using only a simplified nodalization, it must be understood that the simplified model has the potential to produce results with a considerably larger uncertainty than the model and resulting calculation that stem from a user's effort to produce a model with a minimum uncertainty for the purpose of licensing a plant. A model of this kind is typically built by a user who implements the recommended practices and procedures as defined in the system code documentation. Consequently, in using a model, it is advisable to review the background against which the model was originally developed and used.

The calculational objective also should be considered when reviewing the validation efforts that are undertaken to link the results using a particular system code for various IETs with the model for specific nuclear power plant systems. Often, if there has been no attempt to eliminate nodalization inconsistencies between the IET model and the nuclear power plant model when performing validation studies, erroneous conclusions may follow.

4.2.3. Professionalism

If the required quality assurance procedures are not followed to minimize errors in assembling the model, performing the calculations and interpreting the calculational results, significant errors are possible. In the USA, adherence to Appendix B requirements of 10 CFR 50 [5] is required to obtain licensing approval.

4.2.4. Reduction of user effects

A report published by the OECD/NEA–CSNI [37] gives a number of means to reduce the user effect variable in the building and use of system code models. Foremost among the practices for reducing the user effect are:

(a) Use of standard practices and procedures as codified in system code manuals that are specific to the code being used. The approved practices and procedures should include guidance on nodalization for nuclear power plant models, analysis procedures and quality assurance techniques.
(b) Professional training of the user that includes standard exercises and problems.
(c) Presence of experienced mentors for consultation by the new user.
(d) Rigorous quality assurance procedures.

4.3. PLATFORM INDEPENDENCE

Platform independence refers to the problem of using a particular system code on more than one computer system (platform) with little difference in the calculated results from platform to platform (i.e. the final results are not a function of the computer platform). Thus if the same model can be used on several computer platforms with the same version of a system code such that the same calculational result is obtained, the system code is platform independent; however, if the calculational results differ significantly from one system to another, the results are platform dependent.

Generally the results from system codes are platform independent. However, variables that may produce platform dependent effects include:

(a) Extensive round-off error; for example, produced for extensive convergence cycles per time step such as might occur when the calculation proceeds for numerous time steps at or near the saturation line. This situation may arise in combination with the other factors listed below.
(b) Differences in arithmetic operations, such as division operations from one machine to another.

(c) Source code compiler differences.
(d) Faulty programming practices and techniques.

System code programmers make every effort to eliminate item (d); however, this item should be considered in conjunction with item (c).

To ensure that platform dependencies are avoided, some analysts, in particular in the case of licensing calculations, perform all their calculations on the same type of platform; however, this is not required in general.

4.4. QUALIFICATION OF UNCERTAINTY METHODS

A number of different uncertainty methodologies are available for use. Most of the methodologies are described in Ref. [34]: the AEAT method, the CSAU method, the GRS method, the IPSN method and the UMAE method. All of these methods have their advocates and all are used by various organizations to quantify the calculational uncertainty of system codes.

The methods for quantifying the calculational uncertainty of system codes are quite complex. The above methods have been qualified to different degrees using different approaches. Although the calculational uncertainty stemming from any one of the methods cannot be quantified rigorously due to the inherent uncertainties of the process, they can be qualitatively qualified and also compared. A good example is the comparison given in Ref. [37], where the methods are compared using the same SB LOCA data set. The uncertainty method qualification approach was based on:

(a) The same SB LOCA data set that had known instrumentation uncertainties and equipment uncertainties;
(b) Using the best practices and procedures for constructing the appropriate system code models;
(c) Using the best practices and procedures for each uncertainty method.

The results showed some significant variations from one method to another. However, in general, the results produced a new level of confidence in the use of such tools and led to recommendations for further improvements in each process. It is noted that further substantial improvement of calculated uncertainty ranges would require new data to justify new input uncertainty ranges.

Basically no quantitative or qualitative standards exist for ‘qualifying’ the uncertainty methodologies in use today. The uncertainty methodologies accepted for use (and thus considered qualified uncertainty methodologies) share the following characteristics:

(i) The results are reproducible;
(ii) The results are traceable.

In addition, uncertainty methodologies that do not rely heavily on expert panels are generally considered preferable.

5. SUGGESTIONS FOR APPLICATION OF METHODS

The process for performing a BE analysis with uncertainty evaluation using a system code for a selected nuclear power plant transient scenario is relatively straightforward. However, the choices should be considered carefully to ensure that the user does not incur unnecessary costs in terms of time and resources. This section describes the basic steps in the process, the characterization of the scenario, the selection of the code, the nodalization process, the selection of the uncertainty quantification process and the application of the uncertainty process.

5.1. BASIC STEPS FOR PERFORMING AN UNCERTAINTY ANALYSIS

In general, the overall process consists of seven steps: (a) selection of the nuclear power plant and scenario; (b) characterization of the scenario and identification of important phenomena; (c) selection of the code; (d) preparation and qualification of the input deck; (e) selection of the uncertainty method; (f) application of the uncertainty method; and (g) comparison of the results with the relevant criteria.

5.1.1. Selection of nuclear power plant and scenario

This step is usually defined by a need: the need to obtain a licence for an existing nuclear power plant or one under construction, the need to increase the operational power level of an on-line nuclear power plant to enhance its cost–benefit ratio, etc. The scenario is usually defined by either the need to evaluate all limiting scenarios using a best estimate plus uncertainty (BEPU) approach or the need to increase the margin between the calculated and allowable limiting values to enable the relaxation of a nuclear power plant operational limit. Consequently, this step is generally the easiest to define.

5.1.2. Characterization of the scenario and identification of important phenomena

Each scenario to be considered can be subdivided into phases that are a function of the scenario's general behaviour. For example, LB LOCAs are characterized by the blowdown, refill and reflood phases. An SB LOCA transient can be subdivided into phases as well as dominant phenomena.

Although some of the phenomena are readily apparent in each scenario, for example rapid depressurization in the LB LOCA, there are sometimes also numerous additional phenomena that require classification as either dominant, less important or negligible in importance. The phenomena are classified by means of the PIRT process, as described in Section III–1 of Annex III.

5.1.3. Selection of the code

The selection process for the code is also generally easy to accomplish. Factors that contribute to the selection are: availability, applicability and compatibility. The availability of the code normally depends on the cost of obtaining the code and on whether the code is allowed for use in the user's country. Decisive for the applicability of the code is whether it can be used for the desired analysis and whether it is generally acceptable to the user's licensing authorities. The code's compatibility is determined by whether the user can use the suggested system code on the available platforms. The code selection process is outlined in Section 5.3.

5.1.4. Preparation and qualification of the input deck

The input deck should be constructed using the practices and procedures specified in the supporting documentation for the system code in question. For a licensing analysis, the recommended practices and procedures should be strictly followed. They include such items as the input parameters for the model with known and required uncertainties, and the nodalization and applicable correlations, for example the critical flow model. The input deck should be constructed using the quality assurance procedures required for such licensing studies, for example as specified for the USA in Appendix B to 10 CFR 50. Finally, an acceptable steady state calculation and reference calculation should be completed to demonstrate the model's behavioural characteristics and acceptability for the desired calculation.

5.1.5. Selection of the uncertainty method

Assuming that only qualified uncertainty methodologies are considered, the uncertainty methodology to be used is usually dictated by: (a) the requirements of the user's licensing authority; (b) the projected cost of producing the uncertainty calculation; (c) the projected cost of using the methodology; and (d) the projected benefits to nuclear power plant operation.

If the user's licensing authority has reviewed several uncertainty methodologies and pronounced judgement regarding the desirability of one methodology over the others, then this consideration is usually the most important factor in selecting an uncertainty methodology. If several methodologies are plausible, the remaining considerations are generally cost dependent. Items (b), (c) and (d) are direct functions of the cost of developing the methodology for use, using the methodology on an extended basis and/or the ratio of the cost of using a particular methodology to the savings in plant operations that are projected given that the objectives are achieved.

Item (b) follows directly from the cost that will be required to gain acceptance by the licensing authority of the proposed uncertainty methodology. The cost of the acceptance and implementation of a desired approach can be quite high.

Items (c) and (d) follow from the cost of using a methodology relative to the savings in plant operations that are achieved. If the application of a methodology for a plant reload requires extensive modifications to important items such as a response surface that is dependent on the fuel bundle design, and if the process is not automated, then the application cost could be unexpectedly large.

5.1.6. Application of the uncertainty method

The uncertainty method is applied by collecting the appropriate data — as defined for each method — and inserting the data as required by the process. Thereafter, each process should be followed along the lines defined and accepted by the licensing authority.

5.1.7. Comparison with applicable criteria

This final step is defined by the licensing requirements and is generally accomplished by providing such data as the PCTs calculated for an LB LOCA.

5.2. CHARACTERIZATION OF THE SCENARIO

A suitable knowledge and understanding of the transient scenario in question (i.e. of the transient and/or accident assumed to occur at the nuclear power plant concerned) constitutes one of the fundamental requirements of a meaningful uncertainty analysis. Furthermore, the characterization of thermohydraulic scenarios is relevant for code assessment, for the design of experimental facilities, and for the interpretation both of results from system code predictions and of measurements from experiments in ITFs. The last two topics (i.e. interpretation of results from code predictions and of measurements from experiments) in themselves may be considered as 'characterization of the scenarios'.

Reference is made hereafter to transients occurring in LWRs involving changes in fluid and structural material thermohydraulic properties such as flow rate, pressure, density, temperature, exchanged power, etc., in both the primary and secondary sides of reactor systems, as applicable. In addition, working conditions of components such as pumps and valves or of software and connected hardware such as control systems must be modelled in order to characterize transient scenarios. Key issues for the interpretation of a transient scenario are:

(a) Phenomena identification (related to types of transients);
(b) Scenario specific phenomena;
(c) Initial conditions (including ranges);
(d) Imposed sequence of events (including list of assumptions);
(e) Boundary conditions (including ranges);
(f) Time trends and the resulting sequence of main events;
(g) Range of variation of output quantities.

Depending upon the objectives of its characterization (e.g. directed at a single component or a single phenomenon or addressing the entire plant), the characterization of a transient scenario may require a few variables or a number of variables as large as several hundred. In all cases, it is convenient to introduce the concept of phenomenological windows as the first step of the characterization analysis. Phenomenological windows are time periods during a particular transient when a single thermohydraulic phenomenon occurs, or when a limited number of thermohydraulic dominant phenomena occur, or when a single component is mainly responsible for the system performance. Typical examples of phenomenological windows during a complex transient are:

(i) Occurrence of dryout; that is, from the time when the surface temperature of the fuel rods rises above values close to the liquid temperature up to the time when such an event is reversed (i.e. when the fuel rod surface temperature once again approaches the liquid temperature).
(ii) Pressurizer emptying period, as well as steam generator emptying period.
(iii) Time of cycling of the pressurizer valve (or of any safety or relief valve in the system).
(iv) Period when single phase (or two phase) natural circulation is an effective tool to remove thermal power from the core.

One of the main reasons for introducing phenomenological windows into the analysis is the need to simplify the analysis itself by subdividing a complex scenario into more simple parts.

Inside each phenomenological window, phenomena identification (related to types of transients) is carried out. The phenomena are taken from relevant lists provided by the CSNI [6–8, 38]. These phenomena are related to classes of transients and range from 'natural circulation' [6] to 'critical flow' and 'countercurrent flow limiting' [8]. The identification of phenomena, among other things, is necessary to estimate the qualification level of the code adopted in the process under consideration. It must therefore be checked that the phenomenon concerned, including its scaling-up, is within the domain of validation of the adopted code.

The next step is consideration of the scenario specific phenomena. The process and the reason for this step are similar to those in the previous case. In the literature (e.g. Ref. [39]), the scenario specific phenomena are also referred to as relevant RTAs. These are subdivided into:

- Single valued parameters; for example, the peak pressure inside a particular phenomenological window or the minimum mass inventory in the primary system during the entire transient, the overall mass delivered by the accumulators, etc.
- Non-dimensional parameters; for example, the ratio between the core produced power and the thermal power transferred across the steam generator inside a particular phenomenological window.
- Parameters belonging to the time sequence of main events; for example, the time when the fuel rod surface reaches the maximum temperature, the time of actuation of the low pressure emergency system, etc.
- Integral parameters; for example, the overall mass and energy exiting the break, the integral of the core power produced within a phenomenological window, the mass delivered by the accumulators.

Reference [40] gives the minimum number of RTAs necessary to characterize a transient scenario in LWRs as around 40. Knowledge of the scenario specific phenomena (e.g. RTAs) can be interpreted as a synonym for knowledge of the scenario under consideration.

A suitable set of values for the initial conditions (including ranges) is required to begin performing analyses and to characterize the initial state of the plant concerned. Such values typically include pressure, flow rate, fluid temperature, pressure drops, water levels and power in the different parts of the loops. Ranges of variations expected or measured for these conditions (usually not available for each individual scenario) may constitute important sources of uncertainty in the prediction of the transient scenario. The material properties (e.g. thermal conductivity versus temperature) may be considered within the set of initial conditions (or boundary conditions).

A suitable set of imposed sequences of events (including the list of assumptions) is needed to begin performing analyses and to characterize the transient scenarios. A typical imposed sequence of events includes scram signal occurrence, conditions for pump trip, pressure set point for valve opening, conditions for closure of feedwater valves, etc. For each transient, a set of set points for equipment actuation, trips, etc., must be prepared in addition to the associated assumptions listed.

A suitable set of boundary conditions (including ranges) is needed to perform the analyses and to characterize the transient scenarios. Typical boundary conditions are the time of opening or closure of a valve, the core power decay curve, the pump coastdown curve as well as the pump homologous curves.

An applicable figure of merits, including time trends and the resulting sequence of main events, may be seen as a key element for characterizing the transient scenario. Such a figure of merits includes:

— Time trends, for example pressure, rod surface temperature, flow rate at the break, etc. At least 20 time trends are necessary to characterize a complex scenario [39].
— Resulting sequence of main events, for example the time when the pressurizer is empty, the time when scram occurs, etc.
— Identification of RTAs (see above).

Any BE calculation should be characterized by an optimum variation range of output quantities (i.e. the output should show excellent or reasonable agreement with the data (see also Section 4)). These ranges constitute the uncertainty in code predictions or results, and are discussed extensively in this report.

5.3. FLOW CHART FOR SELECTION OF THE CODE

Selecting the code is fundamental to the selection process, which is represented graphically in Fig. 7. A number of system codes have been used together with BEPU methodologies for performing nuclear plant licensing analyses. These codes include, but are not limited to, ATHLET, CATHARE, RELAP5 and TRAC.

Since each system code — used to analyse single and two phase phenomena — was created using a spectrum of correlations and models (such as, for example, flow regime transition models), the region of applicability of each code is linked to the region of applicability of the correlations and the models used to create the code in question. In addition, each of the above codes has been used to various degrees together with BEPU techniques. For example, the CATHARE and RELAP5 codes have both been used in the UMAE process; the RELAP5 and TRAC codes have been used in the CSAU process; and the ATHLET code has been used in the GRS process.

In the event that the code considered has not been used in combination with any of the well known BEPU processes, it is recommended that the decision to use the code be thoroughly reconsidered, since the process of using one of the BEPU processes for the first time in combination with a system code is no trivial matter and will require considerable resources to implement.

The flow chart shown in Fig. 7 assumes that the desired nuclear power plant and scenario have been selected using the approach outlined in Section 5.1. Thereafter the process is defined assuming that the user will choose a system code that has been used in combination with an already qualified uncertainty methodology and according to whether the code has the capabilities to analyse the desired scenario. Generally speaking, when considering Western design nuclear power plants, the correlations and modelling considerations contained in the commonly known system codes have already been shown to be applicable to all the desired scenarios required for licensing calculations.

5.4. PREPARATION OF NODALIZATION

Nodalization of the nuclear power plant model is fundamental to obtaining an acceptable licensing calculation. The proper nodalization procedures are given in the practices and procedures specific to the chosen system code. The nodalization for each component, for example the pressure vessel, pressurizer, steam generators, etc., in the case of PWRs, and the pressure vessel with active components such as jet pumps, in the case of boiling

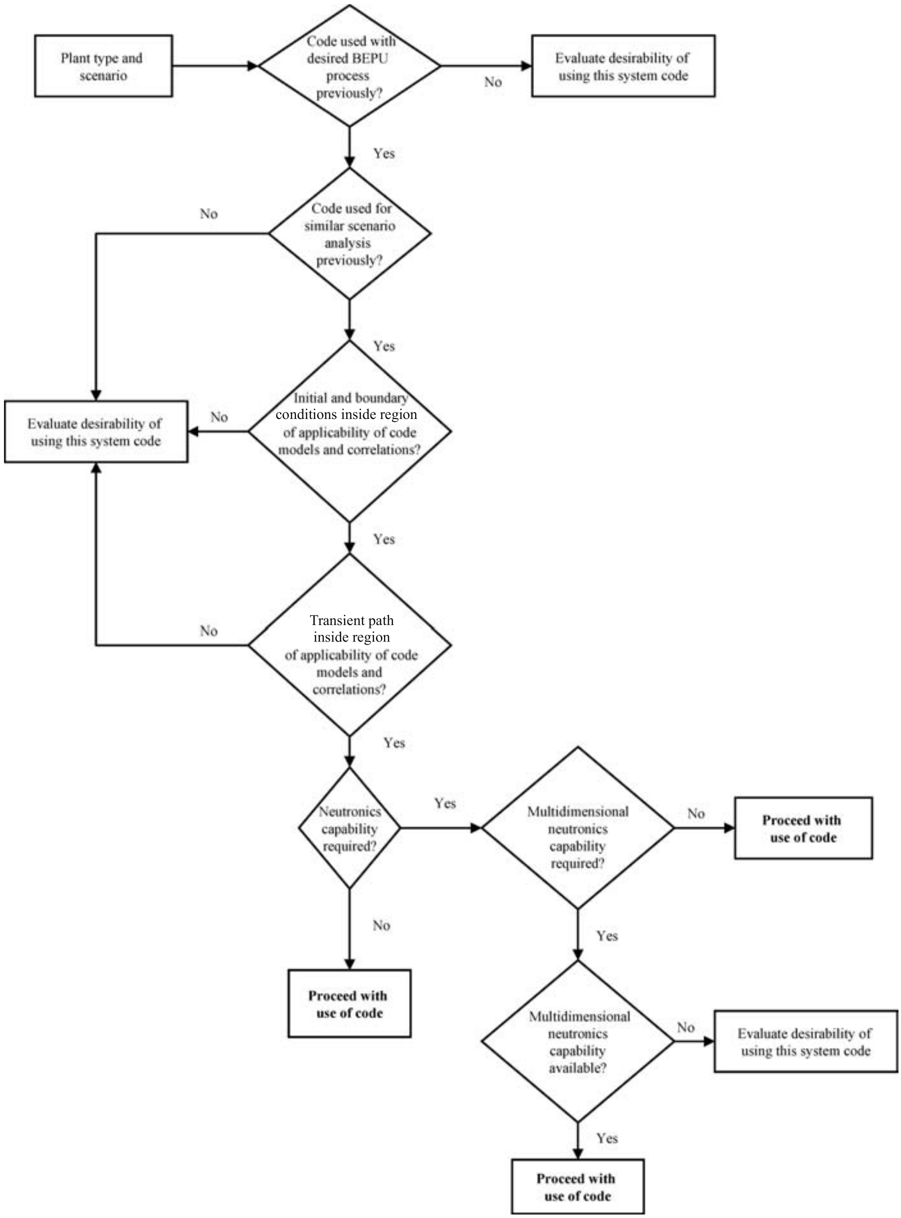

FIG. 7. Flow chart for selecting a system code.

water reactors (BWRs), may be implicitly or explicitly defined in the system code user guidelines. The recommended nodalization practices are based on numerous studies designed to ensure that calculational convergence is achieved with a minimum of cells in all components.

The process of qualifying a system code plant model includes the requirement for ensuring that the nodalization of the plant is consistent with the nodalizations of the integral plant facility models whose data sets were used for meeting the system code validation requirements. The plant model and the integral effects models built to study the related experiments constitute a nodalization validation consistency set. The models used to perform such analyses (i.e. the plant model and the model of the integral effects experiments) are consistent if:

(a) The same modelling practices were used to create the nodalization validation consistency set. Thus, for example, a non-equilibrium two velocity formulation should be used for all models in the set.
(b) The same nodalization practices are used for similar geometries.

The modelling practices should be defined in a practices and procedures document that is specific to the system code that forms the basis for the models. In the event that plant components are not included in the practices and procedures document, new practices and procedures that meet with the approval of the licensing authorities must be developed prior to completion of an acceptable model nodalization.

The objective in creating consistent models is to: (i) minimize the influence of the model nodalization as a variable in the comparative analysis of models in the nodalization validation consistency set; (ii) simplify the comparative analysis process; and (iii) ensure that all the important parameters are considered in the facility transient analyses.

5.4.1. Basis of facility model consistency

Generally speaking, the integral effects facilities that are used to obtain data for validating system codes are constructed using a combination of BE scaling practices and economic considerations. This means that ideally an integral effects facility should be constructed always to produce the best possible scaling relationships between the scaled data and the expected behaviour of the nuclear power plant. However, almost always economic considerations cause hardware approximations to be included such that each facility contains some unique scaling compromises. Consequently, the task of making integral effects models consistent with the nuclear power plant model is often complicated and complex.

The bases for ensuring that the integral effects model nodalizations are consistent with the nuclear power plant model are the following:

(a) The nuclear power plant model is designed to faithfully simulate the dominant or most important phenomena that are expected to occur in the scenario calculation, within the capabilities of the system code that is intended for use. Consequently, the nuclear power plant model is generally the reference and the point of comparison.
(b) The integral effects facility models are constructed to follow the nuclear power plant model practices and procedures to ensure that the same transient phenomena will be represented and that the models will be as similar as possible. To achieve this objective, the integral effects facility models are thus defined to have the same number of cells as the nuclear power plant model whenever practical. However, it is recognized from the start that the integral effects models will differ from the nuclear power plant model as a function of the integral effects facility size, presence of unique components, etc.

5.4.2. Basic rules for nodalization

In every case, established practices and procedures, as defined in the governing documents for the applicable system code, must be used. In the event that the nuclear power plant under consideration has unique components, operational conditions or procedures, engineering judgement must be used to supplement the established practices and procedures until proper validation studies have been completed and agreement with the licensing authorities is achieved.

Guidelines applicable to the nuclear power plant model and its supporting integral effects models fall into two distinct areas: general guidelines and hardware component specific guidelines.

5.4.2.1. General guidelines

General guidelines refer to those peculiar to general thermohydraulic modelling practices, such as, for example, procedures specific to: (a) minor flow paths at arbitrary angles with respect to the major flow path; (b) annuli; (c) modelling interphase drag in complex geometries unique to individual vendor designs; and (d) regions where relatively uniform length to diameter ratios are required.

5.4.2.2. *Hardware component specific guidelines*

Such guidelines refer to nodalization practices applicable to simulating a portion of the system, for example the steam generator or the pressurizer. In general, the component specific practices may be similar to those used elsewhere in the model, but modified to meet special requirements imposed by the hardware design.

5.5. SELECTION OF UNCERTAINTY METHOD

The uncertainty methodologies are inherently difficult to quantitatively qualify since there are no experimental programmes that can be used to conclusively and completely verify the fidelity and accuracy of the process. Hence, it is desirable to perform methodology to methodology comparisons whenever possible. It is assumed that, in the process of carrying out its audit function on such analyses, the applicable licensing authority will either perform comparative calculations or at the very least perform spot checks on key results.

Tables 5 and 6 (based on tables presented in Ref. [17]) compare the five methods from the perspective of the assumptions applied in the methodologies with regard to the characterization of the uncertainties (Table 5) and the ingredients of each methodology (Table 6), respectively. It is helpful to examine the differences between the prominent uncertainty methodologies as an exercise in understanding the differences between them, and thus in understanding the factors that contribute to choosing one methodology over another from a technical perspective.

The attempt to characterize the uncertainties (see Table 5) provides a fundamental understanding of the methodologies. The probabilistic methods — that is, CSAU (the variant of the CSAU framework examined here is ENUSA), GRS and IPSN — all share the same assumed uncertainty characteristics: (a) the values of the quantities of interest are consistent with the available evidence; (b) it is necessary to use PDFs to describe the input uncertainties; and (c) the input uncertainties are described as uniform distributions, particularly when little knowledge regarding these is available. The AEAT and UMAE methods, in contrast, use unique assumptions; for example, the AEAT method projects a reasonable uncertainty range based on the available evidence, while the UMAE method requires that a number of non-statistical conditions be satisfied.

A deeper comprehension is gained by comparing the overall characteristics of the methods (see Table 6). For example, the number of code runs for the probabilistic methodologies is determined by using Wilks' formula; the table shows many similarities between these methodologies, while the AEAT and UMAE methodologies frequently have unique requirements.

A summary of the guidelines for the choice of the most suitable uncertainty analysis method from a technical point of view is provided in Ref. [17]:

> "The following guidance is offered on the selection of uncertainty analysis methods for a particular application:
>
> — If the stringent criteria for the required database and for the accuracy of its modelling by the selected code are fulfilled and if the end users of the uncertainty study can accept the assumptions about extrapolation upon which the method is based… then the Pisa [UMAE] method can be used. This is most likely to be the case when the extrapolation from the data base to the case of interest is small.
>
> — If the end users of the uncertainty study can accept a method based on the combination of probability distributions then a probabilistic method such as the GRS, IPSN, or ENUSA method can be used. When these methods are used, standard tools from statistics can be used to indicate where the state of knowledge needs to be improved so as to improve the knowledge of the predicted quantity most effectively and to form an understanding of the interactions between the important processes. The IPSN method will have the capacity to vary the probability distributions by using regression methods or direct methods.
>
> — If the end users of the uncertainty study prefer not to rely on assigned probability distributions and can rely on the mathematical and physical skill of their analysts to search out the maxima and minima of all the quantities of interest within the uncertainty space then the AEAT method can be used. When this method is used, an understanding of the interactions between the important processes and of the relative importance of the uncertainties in the application being studied (and hence where knowledge improvements are most needed) is built up."

Practically speaking, however, the major factors that usually determine which of the uncertainty methodologies should be used stem from decisions made by the user's licensing authority and from economic considerations, as described in Section 4.4.

TABLE 5. APPROACH TO THE CHARACTERIZATION OF UNCERTAINTIES

Criteria	AEAT	CSAU	GRS	IPSN	UMAE
Useful to know:	What values of a quantity of interest are consistent with the available evidence? Evidence is used to obtain a reasonable uncertainty range that includes all values consistent with the evidence	What values of a quantity of interest are consistent with the available evidence?	What values of a quantity of interest are consistent with the available evidence?	What values of a quantity of interest are consistent with the available evidence?	What values of a quantity of interest are consistent with the available evidence?
Although stochastic quantities, describing the relationship between the measured and calculated quantities, are used, a number of non-statistical conditions must be satisfied before using the parameters					The non-statistical conditions that must be satisfied include the following: the design scaling factors and test design scaling factors must be suitable; sufficient relevant experimental data are available and qualified; the code models must demonstrate the capability to predict the selected experimental data with an accuracy level better than the specified limit; the nodalization and user must be qualified; and the relevant RTAs are the same in the plant calculations and the tests

TABLE 5. APPROACH TO THE CHARACTERIZATION OF UNCERTAINTIES (cont.)

Criteria	AEAT	CSAU	GRS	IPSN	UMAE
State of knowledge about input uncertainties expressed as:		PDFs	PDFs	PDFs	
Minimum knowledge about input uncertainties is expressed as:		Uniform distribution	Uniform distribution	Uniform distribution	

TABLE 6. COMPARISON BETWEEN METHODS

Feature	AEAT	CSAU	GRS	IPSN	UMAE
Characterization of uncertainties	Reasonable uncertainty ranges	Reasonable uncertainty range plus probability distribution	Range of all possible applicable values plus state of knowledge expressed by subject probability distributions	Reasonable uncertainty range plus probability distribution	Accuracy (difference between prediction and measurement) is treated as a stochastic variable

TABLE 6. COMPARISON BETWEEN METHODS (cont.)

Feature	AEAT	CSAU	GRS	IPSN	UMAE
Selection of data	All relevant and available separate effects data should be used	All relevant and available separate effects data should be used	All relevant and available data should be used	All relevant and available data should be used	Must be enough integral experimental data with: suitable facility scaling, suitable test scaling and full range of relevant RTAs
Selection of important uncertainties	Yes	PIRT process	All possibly important uncertainties	All possibly important uncertainties	No
Qualification	ISO 9000-3 or equivalent	Experienced users ISO 9001	Qualified input deck Experienced users ISO 9001	Qualified input deck	Database, nodalization, qualified users Quantitative criteria for prediction of RTAs and qualification of nodalization
Combination and propagation of uncertainties and extrapolation	The analyst explores the multidimensional parameter space for maxima and minima and decides when to stop	Uncertainty space is sampled at random according to the combined probability distribution until a 95% classical statistical confidence limit of the 95% probability content of the calculation result is available	Uncertainty space is sampled at random according to the combined probability distribution until a 95% classical statistical confidence limit of the 95% probability content of the calculation result is available	Uncertainty space is sampled at random according to the combined probability distribution until a 95% classical statistical confidence limit of the 95% probability content of the calculation result is available	Provided that: the data criteria above are fulfilled; the qualification criteria above are fulfilled; the RTAs predicted for plant are in the database; the parameter ranges are properly scaled and are the same; the ratios of measured to calculated values of parameters are randomly distributed about 1:0, such that the measured accuracy of parameters calculated for the database is extrapolated

TABLE 6. COMPARISON BETWEEN METHODS (cont.)

Feature	AEAT	CSAU	GRS	IPSN	UMAE
Number of code runs	Defined by the user	For a two-sided tolerance and confidence interval, Wilks' formula defines the number of runs needed for at least 95% probability content and at least 95% confidence Licensing calculations need one-sided 95%/ 95% tolerance/ confidence limits	For a two-sided tolerance/confidence interval, Wilks' formula defines the number of runs needed for at least 95% probability content and at least 95% confidence Licensing calculations need one-sided 95%/95% tolerance/ confidence limits	For a two-sided tolerance/confidence interval, Wilks' formula defines the number of runs needed for at least 95% probability content and at least 95% confidence Licensing calculations need one-sided 95%/95% tolerance/ confidence limits	One for each test in the database and one for the target transient
Specific data used for scaling	During code validation	During code validation	During code validation	During code validation	Yes
Use of response surfaces to approximate results	No	Yes in classical CSAU; however, not all variants of the CSAU require response surfaces	No	No	No
Use of biases on results	No	No	No	No	Possibly in an identifiable number of situations

TABLE 6. COMPARISON BETWEEN METHODS (cont.)

Feature	AEAT	CSAU	GRS	IPSN	UMAE
Continuous output parameters	Overall maxima and minima of calculated curves or judged envelope curves (if wider)	Not in classical CSAU Variants of CSAU give continuous output similar to the GRS and IPSN methods	By two curves that are the continuous connections of lower and upper endpoints of local 95%/95% tolerance/confidence limits	By two curves that are the continuous connections of the lower and upper endpoints of local 95%/95% tolerance/confidence limits	By linear interpolation
Sensitivity of output parameters to input parameters	Understanding of the processes and their interactions generated	Statistical sensitivity information generated	Statistical sensitivity information generated	Statistical sensitivity information generated	No
Check against independent experimental data from the database used	Yes	Optional	Yes	Optional	Yes

5.6. ADDRESSING RELEVANT STEPS IN SPECIFIC UNCERTAINTY METHODOLOGIES

5.6.1. Selection of input uncertain parameters

5.6.1.1. GRS

First of all, the method requires identification of the potentially important contributors to uncertainty of the code results. These contributors consist of:

(a) Uncertain model parameters;
(b) Uncertain scale effects;
(c) Uncertain initial and boundary conditions (e.g. power, decay heat, etc.);
(d) Uncertain plant parameters (e.g. temperature of ECC water);
(e) Uncertain geometry (e.g. bypass flow cross-sections);
(f) Uncertain fuel parameters (e.g. gap width, fuel conductivity);
(g) Uncertain numerical parameters (e.g. convergence criteria, maximum time step size).

All potentially important uncertainties are selected. There is no need to place any limit on the number of uncertain parameters, since the number of code calculations is independent of the number of uncertain parameters. Parameters that could possibly have an effect on the uncertainty of the code results are included. The decision is based on experience from code validation.

Modelling uncertainties are represented by additional uncertain parameters. These represent two possibilities:

(i) Uncertain corrective additive terms or multipliers;
(ii) A set of alternative model formulations (select between different correlations).

Sets of alternative model formulations can be introduced to select between different correlations, for example from wall heat transfer, and from hydrodynamics (pressure drop, momentum term). In these cases the modelling uncertainty is expressed by uncertain parameters by index numbers of the different correlations. For each individual alternative correlation corrective additive terms or multipliers can be quantified.

It is important that the identified model parameters be independent, and that experience from validation of a specific model by separate effects experiments be addressed by one specific model parameter. If parameters have contributors to their uncertainty in common, the respective states of knowledge are dependent. As a consequence of this, dependence parameter values cannot be combined freely and independently. Instances of such limitations need to be identified and the dependencies need to be quantified, if judged to be potentially important. The quantification is either deterministic (complete dependence) or statistic by measures of association, for example correlation coefficients.

Beyond the model uncertainties, noding changes may be expressed by sets of alternative formulations. Usually, the noding should be considered as optimized based on validation experience and recommendations from the code user's manual.

5.6.1.2. CSAU–PIRT and panel of experts

One of the distinguishing characteristics of the CSAU process is its unique procedure for determining the relative importance of the processes and phenomena that occur during a transient of interest. This is an important aspect of the analysis since there are simply too many processes and phenomena present during LWR SB LOCAs and LB LOCAs to consider them all.

The mechanism used to rank the various processes and phenomena in relation to one another is to use a panel of experts to form a PIRT. When the PIRT process is completed the outcome is a listing, in relative importance, of the 'dominant' phenomena that occur during the scenario of interest. The NRC sponsored the completion of two PIRT studies, one for an LB LOCA using TRAC and one for an SB LOCA using RELAP5.

In general, a PIRT is completed by first assembling a group of experts (from four to eight experts) for the scenario that requires analysis. Whatever scenario is under consideration is subdivided into phases such that each phase can be considered separately, for example the blowdown, refill and reflood phases of an LB LOCA. For each phase the experts list all the phenomena that are known to play a role during the transient and then consider, as a panel, which of the phenomena control the scenario, followed by those phenomena that significantly influence the behaviour of the scenario, followed by those phenomena that play a lesser role. By reaching either consensus or at least a majority opinion, the phenomena are ranked into generally three groups. The controlling phenomena and the phenomena that exert significant influence on the progression and behaviour of the transient are those considered for the CSAU analysis. Phenomena of lesser importance are generally not rigorously analysed.

An alternative to assembling and using an expert panel is to use the analytical hierarchical process (AHP). This process uses probabilities to perform an importance evaluation of the various phenomena and thus fulfil the role of the expert panel.

5.6.2. Assigning a range of variations and/or a probability density function to uncertain parameters

In essence, the ranges of variation were determined by assuming a uniform probability distribution — when either the known distribution is approximately uniform or the distribution is unknown — since equal probabilities or uniform distributions represent the maximum ignorance about the distribution. If more knowledge is available a PDF is used instead of a uniform distribution function.

For appropriate initial values of the parameters of importance, 1σ changes are calculated together with maximum and minimum values — based on the expected transient progression, for example the initial and expected final values — considering the potential maxima and minima extremes that might occur during the transient progression.

5.6.2.1. GRS

For each of the selected individual uncertain parameters (Section 5.6.1.1), probability distributions are specified to quantitatively express the corresponding state of knowledge. This is to account for the fact that evidence from previous code validation or experimental evidence indicates that the appropriate parameter value is more likely to be found in certain subranges of the given range than in others. The probability distribution is called 'subjective' since it expresses the state of knowledge of fixed parameter values rather than stochastic variability. To specify the probability distributions, based on the experience that experts gained from code validation, in-house experts are consulted.

The state of knowledge of the input uncertainties is expressed through PDFs. A probability distribution may be obtained from a sample of measurement values. The 95% quantile, for example, would indicate that 95% of the applicable values of the uncertain parameter lie below this quantile value. However, in thermohydraulic analysis there are many cases in which such frequency data are not available. In these cases PDFs can be uniform, piecewise uniform or of other functional forms. These PDFs are quantitative expressions of the state of knowledge and can be modified if there is new evidence. If suitable observations become available, they can be used consistently to update the PDF.

In probability theory, the uncertainty is characterized as a distribution function that shows the range of values that the actual value may have and what parts of the range the analyst considers more likely than others.

5.6.2.1.1. Selecting the input probability density functions

A key issue in this process is the selection of the input parameter PDFs. This is based on experience from validation of the computer code by comparison between model predictions and test data of integral tests and SETs for the model parameters, as well as on known measurement uncertainties. The noding for the calculations of the experiments and of the uncertainty analysis should be similar, at least for those parameters influencing the uncertainty of the code results most. The main objective is that the selected distribution for each input parameter must fit the analyst's state of knowledge for that parameter. This distribution and its parameters model the reliable and available information about the parameter (the more, the better). The choice of PDF and, more, the evaluation of the uncertainty ranges (in common with all input parameter uncertainty methods, all methods except the UMAE) will affect the calculated uncertainty bands.

If the analyst knows of dependencies between parameters, explicitly multivariate distributions or conditional PDFs may be used.

5.6.2.1.2. Adding new information

New information on any input uncertain parameter may lead the analyst to add uncertain parameters or to change the uncertainty ranges or to change their PDF. This may lead to changes in preferences for the input parameter. That change may cause a logical change in the PDF for the output variable.

5.6.3. Selection of integral test facility experiments for obtaining uncertainty values

More than 30 ITFs have been built and operated so far to simulate the transient scenarios expected in LWRs [6–8, 37, 39, 40]. Altogether, more than 1000 experiments (including shakedown tests) have been performed, and related measurements are available in the sense that they are stored in either paper or electronic format by at least one institution. Availability in this case does not mean that a single group of code users has access to all the databases from all the ITFs; the discussion about actual availability of data goes far beyond the scope of this report.

All of the uncertainty methodologies require, to a different extent and in different steps, the use of ITF data to determine the uncertainty in code predictions. The relevance of ITFs for the 'uncertainty technology' is therefore recognized; all the methods require that thermohydraulic system codes adopted for the uncertainty study be qualified against data obtained from ITFs. A number of (uncertainty) methods require for the uncertainty evaluation analyses of ITF experiments characterized by the thermohydraulic phenomena expected in the nuclear power plant scenario under consideration. However, as already mentioned, ITF data are used in different ways by the different methods. Examples are given below based on the CSAU and UMAE procedures.

5.6.3.1. CSAU

The pioneering nature of the CSAU effort to evaluate uncertainty, and the focus placed in this connection on the characterization of the point value uncertainty in the fuel rod surface peak temperature value, also affect the role of ITFs in this uncertainty method. The systematic use of expert judgement can also be taken as a feature of CSAU in relation to the use of an ITF database. The following provide an idea of the relevance of the ITF database in the CSAU methodology [41]:

(a) Experts participating in the panel discussion for the selection of uncertain parameters should be aware of ITF experiments in the area concerned;
(b) The thermohydraulic codes adopted must have shown their capability in predicting the concerned transient scenarios from the comparison between (ITF) measured and calculated data;
(c) The scaling issue should be addressed with the help of ITF data;
(d) Phenomenological windows and specific transient phenomena are deduced from the analysis of ITF experiments.

It should be noted that there is no direct correlation between data measured in an ITF and uncertainty values predicted by CSAU. However, the existence of relevant ITF data is mandatory for the formulation of engineering judgements in the selection of uncertain parameters and in specifying the range of variations for such parameters.

5.6.3.2. UMAE

The UMAE methodology makes full use of the CSAU recommendations related to the use of experimental (ITF) data and establishes a direct correlation between the capability (by a particular code) to simulate experimental data (such a capability is identified as accuracy) and the error (or uncertainty) in predicting nuclear power plant scenarios. The tight correlation between ITF experiments and UMAE uncertainty predictions can be inferred from the following [11]:

(a) At least three ITF experiments that are similar to the nuclear power plant scenario under consideration must exist in at least three differently scaled ITFs.
(b) The code and the code user must demonstrate their capability to predict the scenarios: a one by one correspondence between RTAs calculated by the code and measured in the experiment must be found, and the quantitative accuracy value (from the comparison between experiments and calculations) must be below an assigned acceptability threshold.
(c) The nuclear power plant nodalization (input deck) concerned must be used to predict at least one of the above mentioned ITF scenarios by performing the so called 'Kv scaled' calculation: RTAs in the calculation and in the experiment must correspond and possible (acceptable) differences must be duly understood (see also Section 5.6.4).
(d) The error in predicting relevant ITF scenarios (under the above conditions) is extrapolated to obtain the uncertainty in code applications to nuclear power plant scenarios.

It should be noted that if no ITF experiments corresponding to the nuclear power plant scenario under consideration have been performed, it is not possible to apply the UMAE and obtain the code uncertainty.

5.6.4. Performing a similarity study

Within the domain of system thermohydraulics, 'performing a similarity study' may have different meanings that are encompassed by the more general term of 'addressing the scaling issue'. This may be done for a specific phenomenon or for a system scenario, as well as for the design of an ITF loop or for the analysis of an experiment. Hereafter, the term 'performing a similarity study' is considered within the framework of an uncertainty evaluation study. In such a case, the performance of a 'similarity study' can refer to a single phenomenon or to the overall system performance. Only the second case is considered here. Therefore, by 'similarity study' the comparison between two sets of data is meant, each one characterizing a transient scenario. Transient scenarios can be the result of a code calculation or of an experiment in an ITF.

Within the UMAE process, similarity studies are foreseen at the following levels [41]:

(a) When comparing the transient scenarios in differently scaled facilities.
(b) When comparing results of the nuclear power plant nodalization applied to the prediction of an experiment in an ITF; in this case, the similarity study is also referred to as the Kv scaled calculation, as mentioned in the previous section.

In both cases, a positive outcome of the study is needed in order to proceed with the application of the method. Guidelines are established for performing the study and criteria are fixed for accepting the results.

In conclusion, a similarity study is recommended by CSAU and some other uncertainty methods, and is mandatory within the UMAE process (see also Section 5.6.3), where strict guidelines are introduced on how to perform such a study and on the acceptability of the results.

5.6.5. Number of code calculations

The number of code calculations is independent of the number of uncertain input parameters and dependent on the tolerance limits' probability content and confidence level, according to Wilks' formula [22, 23] (see Section 3.2.1.2). As a consequence, the number of code runs is independent of the number of selected input uncertainty parameters.

For regulatory purposes, where the margin to licensing criteria is of primary interest, for a one-sided tolerance limit (i.e. for a 95th/95th percentile) 59 calculations should be performed.

5.6.5.1. CSAU

The CSAU demonstration does not use Wilks' formula to determine the number of calculations to be performed to cover the uncertain parameter space. Therefore, the number of calculations is dependent on the number of uncertain parameters p. If one performed a sequential variation of parameter values and selected only a minimum, maximum and nominal (reference) value out of a parameter range, the number of calculations n to be performed would be:

$$n = 2p + 1$$

when no combination of parameters is performed.

In order to cover the whole parameter space one should combine all identified uncertain parameter values. The number of calculations to be performed would then be:

$$n = 3^p$$

This would mean, for example, that one should reduce the number of uncertain parameters to four to limit the number of calculation runs to 81.

Another way is to use response surfaces fitted to a small number of actual code calculation results to substitute the BE computer code and to perform cheaper calculations. Examples of calculation runs for the CSAU demonstration applications are given in Table 3 (i.e. seven calculations for the LB case and eight calculations for the SB case using a BE computer code). A large amount of additional calculations were performed using the fitted response surfaces. The use of response surfaces does allow quantification of point values only, for example the PCT or minimum vessel water inventory, but no time dependent uncertainty quantification.

5.6.5.2. AEAW

It is up to the analyst to perform calculation runs covering the parameter space. The selection of input parameter combinations depends on the analyst's choice; there are no prescribed combinations. The minimum number of calculations should be: 2 × number of parameters + 1 reference calculation. Additional combinations of important parameter values are strongly recommended, and have been used in the AEAW applications (see Table 3).

5.6.5.3. UMAE

Since the deviations of code results from experimental data are quantified to evaluate the uncertainty of computer code models, the number of code calculations is dependent on the number of integral experiments used to complete the comparison process. An additional calculation has to be performed for the transient or accident under investigation. In order to optimize or qualify the nodalization of the plant, several additional calculations are necessary. For an SB loss of coolant demonstration about 20 calculations were performed. It is expected that for a nuclear power plant scenario a few additional code runs are necessary.

5.7. COMPARISON WITH APPLICABLE CRITERIA

As a result of uncertainty analysis, the time behaviour of uncertainty ranges for calculated parameters is obtained. In the correct application of an uncertainty method, any real parameter behaviour should fall, with sufficiently high probability (e.g. 95%), between the lower and upper uncertainty bound, or should be expressed by a one-sided bound (upper or lower). For example, according to the US Code of Federal Regulations, chapter 10, para. 50.46 (a)(1)(i) [5], and NRC Regulatory Guide 1.157 [42], an uncertainty evaluation should make use of probabilistic and statistical methods to determine the code uncertainty. The purpose is to provide assurance that for postulated accidents a given plant will not, with a probability of 95% or higher, exceed the applicable licensing criteria. A statement of this kind, obtained from uncertainty evaluation, is a probabilistic licensing requirement.

Special attention should be devoted to those parameters (or sets of parameters) that are closely related to established acceptance criteria. As an example, typical criteria used in light water design are presented in Tables 7 and 8. In many cases, the comparison with acceptance criteria can be based on directly calculated parameters (pressures, temperatures, departure from nuclear boiling ratio); in others (coolable geometry, pressurized thermal shock (PTS) acceptable doses) this requires more complex considerations. If during the whole process the upper or lower uncertainty bound of the relevant parameters does not exceed the acceptance criteria, the safety requirements are fulfilled.

Rigorous application of an uncertainty method requires that all parameters, or at least those important from the point of view of acceptance criteria, be modelled in a BE fashion and that uncertainty bands be produced

TABLE 7. TYPICAL ACCEPTANCE CRITERIA FOR ANTICIPATED OPERATIONAL OCCURRENCES (TRANSIENTS) OF LIGHT WATER REACTORS

Parameter, component or system	Acceptance criterion
Fuel channel	Departure from nuclear boiling ratio or minimum critical power ratio > 1 with sufficient probability
Fuel pellet	Maximum centreline temperature < T_{melt}
Primary and secondary pressure boundary	Maximum pressure should not exceed the opening pressure of the safety valves

TABLE 8. TYPICAL ACCEPTANCE CRITERIA FOR DESIGN BASIS ACCIDENTS OF LIGHT WATER REACTORS

Applicability	Parameter, component	Acceptance criterion
Reactivity initiated accidents	Fuel pellet	Maximum centreline temperature < T_{melt} Maximum radial average peak pellet enthalpy < limiting value (burnup dependent, fuel design specific)
LOCA	Fuel cladding	PCT (and dryout duration) < 1200°C Maximum local oxidation < 17%
LOCA	Core	Maximum hydrogen generation < 1% of possible value
All DBAs	Core	Long term cooling ensured
All DBAs	Core	Subcriticality margin for safe stable conditions ensured
All DBAs	Pressure boundary	Maximum pressure < 110% of design pressure Maximum temperature < applicable design limit value
LOCA	Containment	Maximum pressure < design pressure Minimum pressure > design pressure
LOCA	Containment differential pressure	Within the given design limits
LOCA	Pressure suppression pool	Maximum allowable temperature, design specific Level swell and pool dynamics
PTS	Reactor pressure vessel	No initiation of a brittle fracture or ductile failure from a postulated defect for all transients and DBAs
DBAs during shutdown, one of barriers (reactor, containment) open	Core	No fuel uncovery in the reactor
All DBAs	Permissible dose (to a member of the public or the plant personnel)	Calculated doses should be below the nationally defined limits for a DBA

for all of them to allow comparison with acceptance criteria. However, in many cases this approach is either not possible (e.g. insufficient knowledge of the phenomena) or not practicable (e.g. very intensive effort required). Up to now, the main attention in uncertainty evaluation has typically been devoted to analysis of the PCT, while other parameters have not been analysed with the same level of detail. For such situations, a simplified approach may apply, in particular if there is a high margin to the acceptance criterion for a given parameter. For example, maximum cladding oxidation can be estimated assuming the upper bound curve for the rod surface temperature (i.e. considering the maximum cladding temperature and the duration of the temperature excursion). Similarly, overall hydrogen production can be estimated assuming the upper bound curve for the rod surface temperature at any axial and radial location of the core

As already indicated, comparison with some acceptance criteria requires more complex consideration. For example, the evaluation of long term coolability requires the use of a coupled primary system and containment code or, in any case, a coupled primary system and containment calculation. In addition, homologous curves are needed for the pumps that recirculate the water between the containment sump and the vessel. Proper consideration should be given to the increase of pressure drop in the pump suction line caused by possible debris accumulation in the containment sump. Evaluation of the energy release to the fluid may entail the need for a coupled three dimensional (3-D) neutronics and system thermohydraulics analysis.

Evaluation of the maximum or minimum pressure in the containment or pressure differences inside the containment requires the use of a coupled primary system/containment code or, in any case, a coupled primary system and containment calculation.

Radioactivity release to the fluid can be estimated in a simplified way through estimation of the number of fuel rods loosing gas tightness. Similar curves to those presented in Fig. 8 can be used where the pressure difference is calculated assuming the lower bound curve for the pressure curve in the primary system and the upper bound curve for the rod surface temperature at any axial and radial position of the core that has been modelled.

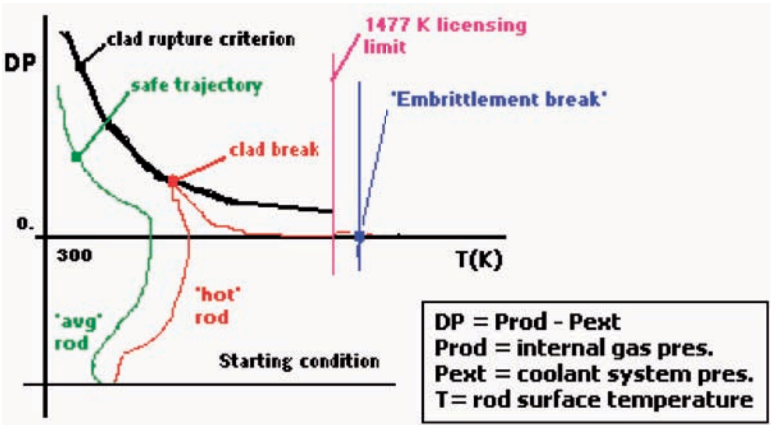

FIG. 8. Qualitative representation of cladding failure conditions during LOCA events.

6. CURRENT TRENDS

6.1. CODE COUPLING

With the advent of increased computing power has come the capability to couple large codes that have been developed to meet specific needs such as 3-D neutronics calculations for partial anticipated transients without scram (ATWS), computational fluid dynamics (CFD) codes to study mixing in 3-D, particularly for passive ECCSs, and others. The range of software packages that are desirable to couple with advanced thermohydraulics system analysis codes include:

(a) Multidimensional neutronics;
(b) Multidimensional CFD;
(c) Containment;
(d) Structural mechanics;
(e) Fuel behaviour;
(f) Radioactivity transport.

There are many techniques for coupling advanced codes. In essence, the coupling may be either loose (meaning the two or more codes only communicate after a number of time steps) or tight such that the codes update one another time step to time step. Whether a loose coupling or a tight coupling is required depends on the phenomena that are being modelled and analysed. For example, the need to consider heat transferred between the primary fluid and the secondary fluid during a relatively slow transient does not require close coupling and thus the codes of interest do not have to communicate time step by time step. In contrast, the behaviour of fluid moving through the core region, where a portion of the core is modelled in great detail using a CFD code while the remainder of the core is modelled using a system analysis code, would require tight coupling if the two codes were linked, since dramatic changes may occur during a nuclear power plant transient. Indeed, since CFD codes generally do not have the capability to model general system behaviour, due to the exceedingly large computer resource requirements, the only means to update a CFD analysis of a somewhat rapid transient in a nuclear power plant core region is via close coupling with a system analysis code that is being used to model the nuclear power plant system. Thus the system analysis code provides boundary conditions to the CFD code if such an analysis need is identified.

There are a number of ways in which two or more codes can be coupled. A description of the various techniques, including their relative advantages, is beyond the scope of this report; however, an example is given below.

6.1.1. Example: RELAP5-3D coupled to FLUENT CF

FLUENT and RELAP5-3D/ATHENA are coupled using a technique that permits implicit interactions between them using an executive program [43]. Hence, if necessary, the executive will allow FLUENT and RELAP5-3D/ATHENA to move forward in calculation space on a time step by time step basis. In addition, the 3-D neutronics subroutine in RELAP5-3D/ATHENA (based on NESTLE) can be used by itself, together with FLUENT, so that a 3-D fluids model can be coupled with a 3-D neutronics model while the balance of the RELAP5-3D/ATHENA subroutines is used to model the system piping and other system components[6].

The executive program uses the parallel virtual machine as the control medium such that the executive program: (a) monitors the calculational progression in each code; (b) determines when each code has converged; (c) governs the information interchanges between the codes; and (d) issues permission to allow each code to progress to the next time step.

[6] This capability is still under development.

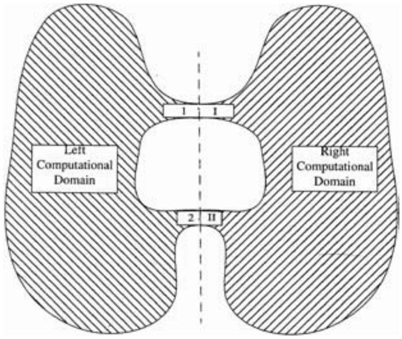

FIG. 9. Schematic diagram of coupled problem solution domain.

The executive program interacts with FLUENT and RELAP5-3D/ATHENA and governs the interactions between FLUENT and RELAP5-3D/ATHENA, since the two codes are each independent domains (see Fig. 9). As noted in Ref. [43]:

> "…volume 1 is adjacent to and connected to volume I, and volume 2 is adjacent to and connected to volume II. The boundary volumes in one of the domains (i.e. 1 and 2) represent normal volumes in the interior of the other computational domain (i.e. I and II). Information about these volumes must be passed between the domains at the coupling boundary to achieve an integrated solution."

Using the above approach, the domains can be coupled explicitly or semi-implicitly, depending on the problem type [43].

While the coupling task described in the previous paragraphs was aimed at providing an analysis tool for advanced reactor systems with a single phase working fluid (gases such as helium, single phase liquids such as lead–bismuth or liquid sodium, etc.), the analysis tool is also applicable to LWR systems, even in the two phase thermodynamic state, for a number of scenarios.

There are a number of phenomena included in the above design characteristics that require the use of CFD codes (see Table 9).

Many phenomena that will require detailed analysis will be present in the advanced supercritical water reactor, the one LWR Generation IV system and the other advanced LWRs for potential accident conditions. Selected phenomena, the events leading to their presence and a concise summary of the concerns that stem from their presence are given in Table 9.

TABLE 9. PHENOMENA COMMON IN ADVANCED REACTOR SYSTEMS UNDERGOING TRANSIENTS REQUIRING COMPUTATIONAL FLUID DYNAMICS ANALYSIS

Required study	Description of phenomena	Typical concerns
Multi-dimensional thermohydraulics in various components	Examples: (i) thermal stratification in suppression tanks during system depressurization; (ii) thermal stratification in inventory make-up tanks; (iii) thermal stratification in movement of fluid plumes in the reactor vessel as cold water is injected into warm water in the presence of a free surface	Studies in thermal stratification and fluid mixing are required to determine whether the fluids adjacent to high temperature components result in unacceptably large thermal stresses, are adequate to prevent thermal failure, etc.
Liquid and gas stratification and interface tracking	Horizontally stratified free surface flow: a common flow regime following depressurization in an advanced reactor system that has both passive injection systems and horizontal pipe runs Passive injection systems lead to free surface flows, since the injection flow rates are not sufficiently large to fill the pipe Consequently, partially filled horizontal pipes lead to large free surfaces, hydraulic jumps, condensation on a free surface, bore flows, stratified countercurrent flow with steam (possibly superheated) moving concurrent with saturated liquid moving countercurrent to subcooled flow	Local overheating or local overcooling
Performance evaluation of passive safety features	Combinations of phenomena described in previous sections that together define the performance of various passive safety systems (i.e. integral system behaviour)	Local overheating or local overcooling

6.1.2. Uncertainty analyses for coupled codes

At present, there are no defined uncertainty methodologies that have been developed and applied to coupled code analyses. Future developments in this area will progress as a function of the coupled software. For example, there are well developed techniques for assessing the uncertainties in some software tools that are at present coupled to system analysis codes, for example multidimensional neutronics software. On the other hand, uncertainty methodologies have not yet been successfully applied to a multiphase problem analysed using a CFD code. Hence their future development for coupled codes is software dependent and is leading edge technology.

6.2. INTERNAL ASSESSMENT OF UNCERTAINTY

All of the uncertainty methodologies suffer from two main limitations:

(a) The resources needed for their application may prove to be prohibitive, up to several person-years;
(b) The results obtained may be strongly methodology user dependent.

The latter item should be considered together with the code user effect, which has been widely studied in the past, as shown in Ref. [44], and may threaten the usefulness or the practical applicability of the results achieved using an uncertainty methodology.

The CIAU approach [18] has been developed bearing in mind the objective of removing the limitations discussed above. Indubitably the 'internal assessment of uncertainty' constitutes a desirable capability of thermohydraulic system codes, allowing the 'automatic' achievement of uncertainty bands associated with any code calculation result. Since a full description of the CIAU method is provided in Annex IV, only an outline of the idea at the basis of the method together with relevant details is presented below.

6.2.1. The idea at the basis of the CIAU

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

(a) Establishment of the nuclear power plant state. Each state is characterized by the value of six relevant quantities (or phases) and by the value of the time since the start of the transient. 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 transient time or duration of the transient scenario is also subdivided into intervals.
(b) Association of uncertainty with the nuclear power plant state. Accuracy values obtained from the analysis of experimental data are associated with each nuclear power plant state.

(c) Use of the method. Each time, the CIAU code calculation result is associated to a time interval and to a hypercube (i.e. a nuclear power plant state), from which the uncertainty values are taken and associated with the current value of the prediction.

6.2.2. General approach and structure

The internal assessment of uncertainty idea can be applied with any uncertainty method. In other words, any existing uncertainty method (CSAU, GRS, etc.) can be used to generate uncertainty values, thus 'filling' the hypercubes and the time intervals.

The idea at the basis of the CIAU is connected with the plant state approach. First, quantities were selected to characterize, in a multidimensional space, the thermohydraulic state of an LWR during any transient. In this way, hypercubes were defined and associated to time intervals accounting for the transient duration. The accuracy of each hypercube and time interval was then calculated from the analysis of experimental data. When applying the method, the combination of accuracy values obtained from hypercubes and time intervals permits continuous uncertainty or error bands to be obtained that envelop any time dependent variables that are the output of a system code calculation.

The RELAP5/MOD3.2 system code and the UMAE uncertainty methodology have been coupled to constitute the CIAU. Therefore the uncertainty is obtained from extrapolation of the accuracy resulting from the comparison between code results and relevant experimental data, which may be obtained from ITFs as well as from SETFs.

A consistent ensemble of uncertainty values is included in any set constituted by a QUM and a TUV. The QUM is formed by hypercubes whose edges are six selected variables representative of a transient scenario. The TUV is formed by time intervals. Four sets of QUM and TUV are included within the framework of the CIAU [18], each set being capable of producing uncertainty bands.

6.2.3. Capabilities of the method

The main advantage of an internal assessment of uncertainty approach, from the methodology user point of view, is that of avoiding having to interpret logical statements that are part of the application process for all the current uncertainty methods; that is, avoiding the user effect when using uncertainty methodologies. This consideration does not exclude the use of engineering judgement; rather, engineering judgement is embedded in the development of the internal assessment of uncertainty method and is not needed in its application. Negligible computer time or human resources are needed for application of the CIAU.

The above characteristics make possible the use of the method for bifurcation studies. The ‘bifurcation possibility’ can be associated with an assigned set of events occurring during the calculation by the CIAU of a nuclear power plant scenario. Each of these events leads to a ‘branch’ in the calculated results. Upper and lower limits can be associated with each branch, ending up in a ‘tree’ of uncertainty bands.

6.3. FACTORS AFFECTING THE REGULATORY PROCESS (LICENSING)

To meet regulatory requirements, safety analysis codes have been used with appropriate conservatism and engineering judgement. A conservative approach has been adopted for licensing analysis, including in the assumptions with respect to plant data, system performance and system availability. Extensive research on the development of computer codes for safety analysis and related experiments has been carried out, providing improved codes and accumulating an extensive database.

In recent years, the industry has made efforts to develop realistic calculation methodologies in safety analysis to improve plant performance, for example through power uprates or by increasing safety margins and reducing the unnecessary conservatism used in safety analysis.

Realistic calculations can be useful for a variety of reasons. Examples are to avoid unnecessary economic penalties, to remove overly restrictive operational practices or improve operational flexibility, to deal with plant ageing effects, and to help resolve outstanding safety issues. Uncertainty assessments are a necessary complement to realistic calculations, providing information on the sensitivity of analysis results to modelling and/or plant variations.

The necessity for the realistic calculation method has not only arisen on the part of the industry but also for the regulator, due to a number of safety issues that are unresolved at present with respect to some low frequency events that traditionally were analysed with extreme conservatism in the past. This has shown that even though licensing analysis appeared to be very conservative, key phenomena and plant and modelling uncertainties have proved to be far more significant than originally estimated.

Some fundamental approaches for the use of the BE method in safety analysis are recommended by the IAEA [4], and it is pointed out that some countries use BE codes and data where practicable so that any cost–benefit assessment of the backfit is not biased by conservatism in the analysis. If BE codes are used, a sensitivity or uncertainty analysis on key parameters (those that are influenced by sensitivities or uncertainties in the plant data, the plant model and the physical model) is recommended to show that there is no large increase in risk if one of these parameters is changed within its uncertainty band. The uncertainty allowance for plant parameters needs to be obtained from operating experience rather than from the values used in the original licensing analysis.

According to 10 CFR 50.46 [5] and NRC Regulatory Guide 1.157 [42], an uncertainty evaluation should, for example, make use of probabilistic and statistical methods to determine the code uncertainty. The purpose is to provide assurance that for postulated accidents a given plant will not, with a probability of 95% (and a 95% confidence level) or more, exceed the applicable licensing criteria. Such a statement, if obtained from uncertainty evaluation, would be a probabilistic uncertainty statement.

The principal regulatory philosophy to achieve the safety goals in nuclear power plant operation is almost identical in all countries, but there are various approaches in the licensing practice. Licensing requirements may be prescribed in detail in some countries, while in others more flexibility may be given to licensees to choose the analysis methods, computer codes and quantitatively evaluated acceptance criteria considered appropriate. However, there should be clear and rigorous guidelines for the use of the BE method with uncertainty quantification. Some examples of its application for licensing are described in Annex V.

It is necessary to incorporate into the licensing process the regulatory basis for the use of a realistic calculation method in the safety analysis. As a first step, trial applications of a realistic calculation are useful for the regulator to assess the feasibility of moving towards a more methodical use of realistic calculations in licensing. Such methods can become an established part of the licensing framework and guidelines. Apart from the trial licensing application, the industry can take a leading role in developing the methodology for its use in safety analysis and to validate it. In parallel, the industry could interact with the regulator so as to move in a common direction.

Some key issues when developing the framework for the use of the BE methodology are:

(a) The potential complexity of an analysis methodology;
(b) The adequacy of the underlying analytical techniques and of computer code validation;
(c) The method by which uncertainties are combined;

(d) The potential need for a stronger tie between plant operation and the analysis;
(e) The degree of statistical rigour required;
(f) The need for good quality documentation.

The regulatory guideline should describe a process for the development and assessment of the evaluation models that can be used to analyse transient and accident behaviour. It should also provide guidance on realistic accident analyses, thus affording a more reliable framework for estimating the uncertainty when investigating transient and accident behaviour. The fundamental features of accident analysis methods are the evaluation model concept and the basic principles important for the development, assessment and review of these methods.

Some major elements that will be described in the regulatory guidelines are:

(i) Requirements and capabilities:
 - To determine the scenario modelling requirements and compare them with the computer code's capabilities in order to establish the applicability of the computer code to a particular scenario.
 - To identify potential limitations.

(ii) Assessment and ranking of parameters:
 - To determine the effects of the important parameter on the specific ranges. The effects to consider include those associated with code accuracy, effect of scale and parameter range for the uncertainty evaluation.

(iii) Sensitivity and uncertainty analysis:
 - The effects of individual contributors to the total uncertainty are determined and combined to provide a statement on the total uncertainty of the analysis.
 - To allow probabilistic uncertainty.

7. CONCLUSIONS

The BE calculation results from complex thermohydraulic system codes are affected by approximations that are unpredictable without the use of computational tools that account for the various sources of uncertainty. In a general case, when conservative input conditions are adopted, the conservatism in the results cannot be ensured because of the obscuring influence that an assigned input conservative parameter value may have upon the prediction of the wide variety of phenomena that combine typical reactor accident scenarios. In addition, the amount of conservatism, when this can be ensured for an assigned output quantity, may suffer from two limitations: (a) it does not correspond to a conservatism in the prediction of a different system relevant variable (e.g. a conservative prediction for rod surface temperature does not correspond to a conservative prediction of emergency system flow rate or of containment pressure); and (b) the amount of conservatism is unknown.

A review of existing uncertainty methods has been accomplished in this report, making reference to the BE prediction of nuclear power plant accident scenarios. Sources of uncertainties and significant features of the uncertainty methods, as well as significant results from their application, have been described.

The pioneering role in this area by the CSAU framework developers at the beginning of the 1990s, and its first application, is recognized. Their work formed the bases for the development of a number of uncertainty methodologies in which the CSAU framework requirements were considered and embedded into methodologies that are less dependent upon expert judgement than the original application.

Uncertainty quantification methods are available today, and several applications have been and will be performed in reactor safety research as well as in licensing. Experience from applications shows that the difference between predicted upper bound or 95th percentile and 95% confidence level PCT and a calculation using nominal BE input values and default values for the computer code options and input data for models (reference calculation) is about 200 K for a typical LB LOCA (see Annexes I and II). These relatively large values are due to the numerous models and correlations that are incorporated into a thermohydraulic code and to the uncertainties associated with the individual models.

Two broad classes of uncertainty methods have been identified dealing with propagation of 'input uncertainties' and of 'output uncertainties'[7], respectively. In the former class, deterministic and probabilistic approaches have been distinguished.

The main characteristics of the methods based upon the propagation of input uncertainties are that they identify the number of input uncertain parameters, assign probability distributions and propagate the uncertainty by

[7] Propagation of 'output uncertainties' is also characterized as extrapolation of output errors.

performing code calculations, which, by their nature, are approximations of the physical behaviour. The main characteristics of the methods based upon the propagation of output uncertainties are the need to have relevant experimental data available and the process of error extrapolation that is not supported by theoretic formulations.

The use of engineering judgement in the development of the uncertainty methodologies and the specification of expert evaluation in their application (in some cases) allow the resolution of the above drawbacks, as proved by the qualification results for the methodologies. It was found that independent principles are the basis of the methodologies in the two classes.

As a main conclusion from the present effort, it is clear that the qualified uncertainty methodologies are maturing. It is recognized, however, that the foremost factor in the promotion and use of the various uncertainty methodologies is acceptance by the governing nuclear power plant licensing authorities in the various countries. It is also recognized that the process of gaining approval from the appropriate licensing bodies to use any of the uncertainty methodologies can consume considerable resources and can require substantial time. Hence it is important that these factors be considered when proposing the use of a given uncertainty methodology.

The introduction of BE methods with uncertainty quantification into the licensing framework was not considered in the present activity and remains an objective to be pursued.

7.1. RECOMMENDATIONS

The use of BE applications of complex thermohydraulic system codes, supported by uncertainty evaluation of the relevant output quantities, is recommended as a means of providing a better understanding of safety margins.

The internal assessment of uncertainty is a desirable capability in the area identified by the technical community in 1996, since it allows the automatic association of uncertainty bands to code calculations results, where uncertainty is a ‘peculiarity’ of a particular code. The influence of the code user upon the predicted uncertainty values should be negligible when a robust method is available. The recommendation here is to explore this area taking into consideration the economic benefit of internal assessment of uncertainty applications.

It is recommended that consistent procedures for the application of uncertainty methods within the licensing process be developed.

Fundamentally, no quantitative or qualitative standards exist for qualifying the uncertainty methodologies in use today. The uncertainty methodologies accepted for use share the following characteristics:

(a) The results are reproducible;
(b) The results are traceable.

Users of the system codes should be properly trained prior to performing plant safety analysis.

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Annex I

SOURCES OF UNCERTAINTIES

The application of BE (realistic) computer codes to the safety analysis of nuclear power plants involves the evaluation of uncertainties. This is connected to the approximate nature of the codes and of the process of code application. In other words, 'sources of uncertainty' affect predictions by BE codes and must be taken into account.

Three major sources of uncertainty are mentioned in annex II of the IAEA Safety Report on Accident Analyses for Nuclear Power Plants [I–1]:

(a) Code or model uncertainty;
(b) Representation or simulation uncertainty;
(c) Plant uncertainty.

A more detailed list of uncertainty sources can be found in Ref. [I–2], in which an attempt was been made to distinguish independent sources of basic uncertainty. The list includes the following items:

(i) 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 is given to geometric discontinuities, which is an uncommon situation for code applications to the analysis of nuclear power plant transient scenarios.

(ii) Presence of different fields of the same phase, for example liquid droplets and film. Only one velocity per phase is considered by the codes, thus resulting in another source of uncertainty.

(iii) Geometry averaging at a cross-section scale. The need to average the fluid conditions at the geometry level makes the porous media approach necessary. Velocity profiles happen in 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'.

(iv) Geometry averaging at a volume scale. Only one velocity vector (per phase) is associated with a hydraulic mesh along its axis. Different velocity vectors may occur in reality, for example inside the lower plenum of a typical reactor pressure vessel or at the connection between the cold leg and the downcomer. The volume averaging constitutes a further uncertainty source of 'geometric origin'.

(v) Presence of large and small vortices or eddies. 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), which may not necessarily be consistent with the prediction of a code discretized model.

(vi) The second principle of thermodynamics is not necessarily fulfilled by codes. Irreversible processes occur as a consequence of accidents in nuclear reactor systems. This causes energy degradation (i.e. transformation of kinetic energy into heat). Calculation of the amount of energy transformation is not necessarily within the capabilities of current codes, thus representing a further specific energy source.

(vii) Models of current interest for thermohydraulic system codes comprise a set of partial derivative 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.

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

- The ranges of validity are not fully specified. For example, pressure and flow rate 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 incorporated into the code, the correlations are applied to situations where, for example, 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 the reactor pressure vessel: no facility has been used to investigate (or to qualify) friction factors in two phase conditions when pipe diameters are of the order of one metre. In addition, once the correlations have been incorporated into the code, no (automatic) action is taken to check whether the boundaries of validity (i.e. those that have been assigned) are exceeded during a specific application.
- Correlations are implemented approximately in the code. The correlations, apart from special cases, are developed by scientists or in laboratories that are not necessarily aware of the characteristics or of the structure of the system code into which 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 the consequences thereof not always ascertained.

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

(ix) A paradox should be noted: a steady state and fully developed flow condition is a necessary prerequisite or condition adopted when deriving correlations. In other words, all qualified correlations must be obtained under steady state and fully developed flow conditions. However, almost in no region of the nuclear power plant do these conditions apply during the course of an accident.

(x) The state and the material properties are approximate. Various materials used in a nuclear power plant are considered in the input deck, including liquids, gases and solids. Thermophysical properties are part of the codes or constitute specific code user input data. These are of an empirical nature and are typically subjected to the limitations discussed under (viii). A specific problem within the current context can be associated with the derivatives of the water properties.

(xi) Code user effect exists. Different groups of users using the same code and with the same information available for modelling a nuclear power plant do not achieve the same results. The user effect originates from:

- Development of the nodalization (see also (xiv));
- Interpretation of the supplied (or available) information, which is usually incomplete (see also (xiii));
- 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 'acceptable' solution.

The user effect may prove to make the largest contribution to the uncertainty and is connected with user expertise and the quality and comprehensiveness of the code user manual and of the database available for performing the analysis.

(xii) Computer and 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. The 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 current experience has shown 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.

(xiii) Imperfect knowledge of boundary and initial conditions. Some boundary and initial condition values are unknown or only approximately known; the code user must add information. This process inevitably causes an impact on the results that is not easily traceable and constitutes a specific source of uncertainty.

(xiv) Code model deficiencies cannot be excluded. System code development started towards the end of the 1960s and systematic assessment procedures have been available since the 1980s. 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 today. Examples of phenomena for which some thermohydraulic system code models prove deficient are:

- 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 flowing down inside a steam–gas filled region.

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

All the above sources of uncertainty are quite well understood by the technical and scientific community dealing with system code development and application. Complex interactions among the basic uncertainty sources are to be expected, and justify the complex structure of an uncertainty method.

Comprehensive research programmes have been completed or are in progress aimed at the assessment and improvement of thermohydraulic system codes to reduce the influence of basic uncertainties upon the results. Difficulties arising from this process are outlined below.

The code assessment process emphasizes differences between predicted and experimental data that cannot be directly or easily assigned to any of the above listed categories. In addition, improvement in the capability of the code to predict a particular experiment does not imply improvement of the capability to predict a different experiment. The process of code assessment improvement is definitely a lengthy one that cannot be expected to fully eliminate the effect of any of the outlined sources of uncertainty. This again substantiates the need for uncertainty studies to be associated with BE calculations.

In Sections I–1 to I–5, the three broad sources of uncertainty indicated in Ref. [I–1] are considered and are supplemented by two additional sources.

I–1. CODE UNCERTAINTY

A thermohydraulic system code 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 three sets of balance equations deal with the fluids of the system, the solid structures including the fuel rods, and the neutrons. The fundamental principles of thermodynamics (namely the first principle) and of mechanics (namely the Newton principle) are at the basis of these equations. In relation to the fluids of the system, the 1-D UVUT (unequal velocities, unequal temperatures) set of partial differential equations is part of the codes under consideration. (Some codes have 3-D UVUT capability, but the related assessment improvement process cannot be considered as completed.) The 1-D Fourier model is solved within the solid structures and is coupled with the fluid balance equations also through the heat transfer coefficient correlations, as described below. (Some codes adopt a 2-D Fourier equation to calculate the reflood phenomenon in the core region.) The 0-D or point neutron kinetics model is part of the codes under consideration. (It is now common to couple 3-D neutron kinetics codes with the thermohydraulic system codes.) It is coupled with the 1-D UVUT and the 1-D Fourier equation also through the average moderator density and the average fuel temperature.

The closure (constitutive) equations deal with the interaction between the fluid and the environment as well as with the interaction of the two phases of the fluid (i.e. the gas and the liquid phase). The interfacial drag coefficient, wall to fluid friction factor and heat transfer coefficient are typically expressed by constitutive equations.

Typically, various sets of materials properties are embedded into the codes, even though the user may change these properties or add new materials. Water, nitrogen, air, uranium dioxide, stainless and carbon steel and zircaloy are materials the thermophysical properties of which are part of the thermohydraulic system codes. Different levels of sophistication usually characterize the sets of properties in the different codes. This is especially true for water (Mollier diagram quantities and related derivatives).

Balance equations are not sophisticated enough for application in the modelling of special components or for the simulation of special processes. Examples of these components are the pumps and the steam separators, and examples of the special processes are the countercurrent flow limiting condition and two phase critical flow, although this is not true for all the codes. Empirical models ‘substitute’ the balance equations in such cases.

The entire set of equations that can be obtained from the models outlined above must be coupled with a numerical solution method that allows the values of the unknowns to be determined at each time step during an assigned transient.

The sources of uncertainty connected with the code are those identified under (i–x) and (xiv) in the list provided at the beginning of this annex. The following association between uncertainty sources and code parts applies:

(a) Balance equations: uncertainty sources (i–vi).
(b) Closure and constitutive equations: uncertainty sources (viii and x).
(c) Material properties: uncertainty source (x).
(d) Special process and component models: uncertainty sources (viii, x and xiv).
(e) Numerics: uncertainty source (vii).

I–2. REPRESENTATION UNCERTAINTY

Representation uncertainty is related to the process of setting up the nodalization (idealization). The nodalization constitutes the connection between the code and the 'physical reality' that is the subject of the simulation. The process for setting up the nodalization is an activity carried out by a group of code users that aims at transferring the information from the real system (e.g. the nuclear power plant), including the related boundary and initial conditions, into a form understandable to the code. Limitation of available resources (in terms of person-months), lack of data, the target of the code application, capabilities or power of the available computational machines and expertise of the users play a role in this process. The result of the process (the logical steps are outlined in greater detail in Section 5.4) may strongly affect the response of the code.

I–3. SCALING

Scaling is a broad term used in nuclear reactor technology as well as in basic fluid dynamics and in thermohydraulics. In general terms, scaling indicates the need for the process of transferring information from a model to a prototype. The model and the prototype are typically characterized by different geometric dimensions, but thermohydraulic quantities such as pressure, temperature and velocities may be different in the model and in the prototype, as well as in the materials adopted, including working fluids.

Therefore, the word ‘scaling’ may have different meanings in different contexts. In system thermohydraulics, a scaling process, based upon suitable physical principles, aims at establishing a correlation between phenomena expected in a nuclear power plant transient scenario and phenomena measured in smaller scale facilities or phenomena predicted by numerical tools qualified against experiments performed in small scale facilities.

Owing to limitations of the fundamental equations at the basis of system codes, the scaling issue may constitute an important source of uncertainties in code applications and may envelop various individual uncertainties. Referring to the list identified previously, the sources of uncertainty connected with the scaling are those applicable to the balance equations, for example those identified under (i–x). More precisely, the uncertainty sources associated with scaling are (i–v) and (viii and x).

The uncertainty associated with scaling may be attributed to the insufficiently ‘uncertainty driven’ code assessment process.

I–4. PLANT UNCERTAINTY

Uncertainty or limited knowledge of boundary and initial conditions and related values for a particular nuclear power plant are referred to as plant uncertainty. Typical examples are the pressurizer level at the start of the transient, the thickness of the gap of the fuel rod, the conductivity of the UO_2, and the gap between the pellets and the cladding.

It should be noted that quantities such as gap conductivity and thickness are relevant for the prediction of safety parameters (e.g. the PCT) and are affected by other parameters such as burnup, knowledge about which is not as detailed as required (e.g. knowledge about each layer of a fuel element that may be part of the nodalization). Thus a source of error of this kind in the class of ‘plant uncertainty’ cannot be avoided and should be accounted for by the uncertainty method.

The source of uncertainty connected with the plant is identified under (xiv) in the list.

I–5. USER EFFECT

Complex system codes such as ATHLET, CATHARE, RELAP5 and TRAC have many degrees of freedom that lead to misapplication (e.g. not using the countercurrent flow limiting model at a junction where it is required) and errors by users (e.g. inputting an incorrect length of a system component).

In addition, even two competent users will not approach the analysis of a problem in the same way and are therefore likely to take different paths to reach a solution. The cumulative effect of user community members to produce a range of answers for a well defined problem with rigorously specified boundary and initial conditions is called the user effect.

To reduce the user effect, several features are required:

(a) Misapplication of the system code should be eliminated (or at least reduced) by means of a sufficiently detailed code description and by relevant code user guidelines.
(b) Errors should be minimized: any analysis of merit should include quality assurance procedures designed to minimize or eliminate errors. In a sense, the misapplication of the system code is itself a certain class of error.
(c) The user community should preferably use the same computing platform. This means, for example, that the machine round-off errors and treatment of arithmetic operations are assumed to be the same.
(d) The system code should preferably be used by a relatively large user community (a large sample size).
(e) The problem to be analysed should be rigorously specified; that is, all geometrical dimensions are unambiguously defined, the initial conditions and boundary conditions are clearly specified, etc.

Within the defined framework, the user effect can be quantified and is a function of:

(i) The flexibility of the system code;
(ii) The practices used to define the nodalization and to ensure that a convergent solution is achieved.

The flexibility of the system codes under consideration is a primary reason for generating a user effect. An example is the flexibility associated with modelling a system component such as the steam generator. For example, the TRAC code has a specific component designed to model steam generators, whereas a steam generator model created using RELAP5 is constructed of basic model components such as PIPE, BRANCH, etc. Consequently, there are more degrees of freedom available to the user, which each require a decision, when a RELAP5 steam generator model is being constructed than when a TRAC generated model of the same component is being defined. As a result, the RELAP5 results for this particular case will have a larger user effect than the TRAC results.

The influence of different users of the same version of a thermohydraulic computer code on the calculated results is partially described in Section 4.2, which discusses the reasons for and the best practices to reduce this effect. The effect of the user on a code result should be minimized as much as possible. Some of the user effects result from imprecise knowledge of the appropriate parameter value or an inappropriate choice of code model. These uncertainties are to be quantified by the uncertainty ranges and/or probability distributions of uncertain input parameters. They should be determined assuming a very experienced user. The possible choices of an inexperienced user should not be taken into account. This applies for uncertainty methods propagating input uncertainties (see Section 3.2).

The impact of the user effect upon the final result (i.e. BE prediction plus uncertainty) may be different depending upon the selected uncertainty method. For methods extrapolating the output error, calculation comparisons of experimental data with the code results are performed to obtain satisfactory agreement between corresponding measured and calculated data (according to established qualitative and quantitative acceptability criteria and thresholds). ITF nodalizations must be developed that are similar, to the extent possible, to a reactor nodalization. The intention is to reduce the user effect as far as possible.

The sources of uncertainty connected with the code user are those identified under (xi and x). The code user bears part of the responsibility associated with the source of uncertainty specified under (xii).

REFERENCES TO ANNEX I

[I–1] INTERNATIONAL ATOMIC ENERGY AGENCY, Accident Analysis for Nuclear Power Plants, Safety Reports Series No. 23, IAEA, Vienna (2002).

[I–2] WICKETT, T., et al., Report of the Uncertainty Methods Study for Advanced Best Estimate Thermal Hydraulic Code Applications, 2 vols, Rep. NEA/CSNI R(97)35, OECD, Paris (1998).

Annex II

DESCRIPTION OF METHODS AND EXAMPLES OF RESULTS

II–1. GRS

II–1.1. Application to the LSTF-SB-CL-18 experiment

II–1.1.1. Description of test

The test simulates an SB LOCA (5%) performed on the Japanese Large Scale Test Facility (LSTF), which is a 1/48 volumetrically scaled model of a Westinghouse type 3423 MW(th) four-loop PWR [II–1]. The main components of the LSTF have the same elevations as the reference PWR, to simulate the natural circulation phenomena, and large loop pipes to simulate the two phase flow regimes and phenomena of significance in an actual plant. The four primary loops of the reference PWR are represented by two loops of equal volume (the inlet diameter is 0.207 m).

Both the initial steady state conditions and the test procedures were designed to minimize the effects of LSTF scaling compromises on the transient during the test. The main operational conditions are:

(a) Break opening at time zero;
(b) Loss of off-site power at scram;
(c) High pressure safety injection not actuated;
(d) Main feedwater termination at reactor scram;
(e) Auxiliary feedwater not actuated;
(f) Accumulator injection at 4.51 MPa;
(g) Lower pressure injection at 1.29 MPa.

The main physical phenomena observed during this test were two uncoveries of the heater rod bundle representing the core. The first one was due to water level depression (120–155 s) before the loop seal cleared (140 s), and the second one (420–540 s) was due to loss of water inventory at the break, which was finished by the accumulator injection (455 s). The whole transient lasted 900 s.

II–1.1.2. Uncertain parameters

All potentially important uncertain parameters are included in the uncertainty analysis. Table II–1 lists the selected input parameters and their specified ranges and distribution types. Included are 41 model parameters, four uncertainties of bypass flow cross-sections in the test vessel, one uncertain heater power and two uncertain convergence criteria of the code. The quantification of model uncertainties is based on the experience gained from validation of the ATHLET code.

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS

No.	Parameter	Ranges Min.	Ranges Max.	Reference	Distribution	Explanation
Critical break flow						
1	DSCON	0.5	3	1.3	Polygonal	Correction factor contraction length
2	FD	0.02	0.22	0.02	Polygonal	Weisbach–Darcy wall friction coefficient
3	FF	0.7	1	0.775	Polygonal	Contraction coefficient for steam flow
4	PP	0.98	0.999	0.98	Polygonal	Transition of void fraction for contraction coefficient
Evaporation						
5	ZBO	10^8	10^{10}	5×10^9	Polygonal	Number of bubbles per unit volume (m^{-3})
6	ZT	10^8	10^{10}	5×10^9	Polygonal	Number of droplets per volume (m^{-3})
7	OMTCON	0.5	2	1	Uniform	Direct condensation
8	TURB	1	50	20	Log-normal	Turbulence factor for evaporation in critical break flow model
Drift models						
9	ODVRO	0.5	1.5	1	Polygonal	Correction factor for vertical pipe (flooding based drift flux model)
10	ODBUN	0.3	1.5	1	Normal	Correction factor for vertical bundle
11	ODVKU	0.7	1.3	1	Normal	Correction factor for vertical annulus

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS (cont.)

No.	Parameter	Ranges Min.	Ranges Max.	Reference	Distribution	Explanation
12	ODHPI	0.75	2.25	1	Polygonal	Correction factor for horizontal pipe
13	ODHBR	0.5	2	1	Uniform	Correction factor for horizontal core channel connections
14	ODENT	1	3	1	Uniform	Correction factor for water entrainment in vertical bundle
Two phase pressure drop						
15	ITMPO	1 or 4				Correlation selection (parameters 16 and 17)
16	OFI2H	1			Log-normal	Martinelli–Nelson correlation with constant friction factor, horizontal (ITMPO = 1)
					Log-normal	Chisholm correlation with calculated friction using wall roughness, horizontal (ITMPO = 4)
17	OFI2	1			Log-normal	Martinelli–Nelson correlation with constant friction factor, vertical (ITMPO = 1)
					Log-normal	Chisholm correlation with calculated friction using wall roughness, vertical (ITMPO = 4)
Pressure drop, wall friction						
18	ALAMO	0.01	0.03	0.02	Triangular	Pipe wall friction (option ITMPO = 1)
19	ALAMO	0.01	0.03	0.02	Triangular	Rod bundle wall friction (option ITMPO = 1)
20	ROUO	10^{-5}	10^{-4}		Polygonal	Pipe wall roughness (option ITMPO = 4)
21	ROUO	1.5×10^{-6}	2×10^{-5}		Polygonal	Rod bundle wall roughness (option ITMPO = 4)

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS (cont.)

No.	Parameter	Ranges Min.	Ranges Max.	Reference	Distribution	Explanation
Main coolant pump						
22	YHS	Table	Table	Table	Uniform	Two phase multiplier for head and torque
Bypass flow paths						
23	CSA	0.01	0.6	0.47	Uniform	Correction factor for bypass flow cross-section between upper downcomer and upper plenum
24	CSA	0.2	1	0.62	Uniform	Correction factor for bypass flow cross-section between upper downcomer and upper head
25	ZFFJ0/ ZFBJ0	0.4	2.5	1	Uniform	Correction factor for bypass form loss between rod bundle and upper head
26	ZFFJ0/ ZFBJ0	0.33	3	1	Uniform	Correction factor for bypass form loss between upper plenum and upper head
Pressure drop, momentum term						
27	JDPA				0.25	Momentum term hot leg and upper plenum from hot leg only (25%)
	JDPA				0.25	Momentum term hot leg and upper plenum not computed (25%)
	JDPA				0.5	Momentum term hot leg and upper plenum in both directions (50%)
28	JDPA				0.25	Momentum term cold leg and downcomer from cold leg only (25%)
	JDPA				0.25	Momentum term cold leg and downcomer not computed (25%)

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS (cont.)

No.	Parameter	Ranges		Reference	Distribution	Explanation
		Min.	Max.			
	JDPA				0.5	Momentum term cold leg and downcomer in both directions (50%)
29	JDPA				0.5	Momentum term at heater rod bundle inlet not computed (50%)
	JDPA				0.5	Momentum term at heater rod bundle inlet in both directions (50%)
Pressure drop, form losses						
30	ZFFJ0/ ZFBJ0	0.667	1.5	1	Uniform	Correction factor for form loss at branch
31	ZFFJ0/ ZFBJ0	0.5	2	1	Uniform	Correction factor for form loss at upper bundle plate and spacers
32	ZFFJ0/ ZFBJ0	0.4	2.5	1	Uniform	Correction factor for form loss at downcomer cross-connections
33	ZFFJ0/ ZFBJ0	0.8	1.25	1	Uniform	Correction factor for form loss in surge line
Heat transfer						
34	IHTCI0	1 or 2				Selection of correlation (parameter 35)
35	OHWFB	0.65	1.3	1	Uniform	Correction factor for film boiling, modified Dougall–Rohsenow correlation (50%)
		0.75	1.25	1	Polygonal	Correction factor for film boiling, Condie–Bengston (50%)
36	ICHFI0	0 or 4				Selection of correlation (parameter 37)

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS (cont.)

No.	Parameter	Ranges Min.	Ranges Max.	Reference	Distribution	Explanation
37	OTRNB	0.7	1.3	1	Uniform	Correction factor for critical heat flux, minimum value (50%)
		0.7	1.3	1	Uniform	Correction factor for critical heat flux, Biasi correlation (50%)
38	OHWFC	0.85	1.15	1	Uniform	Correction factor for single phase forced convection to water (Dittus–Boelter)
39	OHWNC	0.85	1.15	1	Uniform	Correction factor for single phase natural convection to water (McAdams)
40	IHTC30	1 or 2				Selection of correlation (parameter 41)
41	OHVFC	0.8	1.2	1	Uniform	Correction factor for single phase forced convection to steam (Dittus–Boelter II, 50%)
		0.85	1.25	1	Uniform	Correction factor for single phase forced convection to steam (McEligot, 50%)
42	OHWNB	0.8	1.2	1	Uniform	Correction factor for nucleate boiling (modified Chen correlation)
43	OHWPB	0.75	1.25	1	Uniform	Correction factor for pool film boiling at natural convection (Bromley correlation)
44	OTMFB	0.9	1.28	1	Uniform	Correction factor for minimum film boiling temperature (Groeneveld–Stewart correlation)
45	HECU/ HTCLO	20	100	50	Uniform	Accumulator heat transfer coefficient (W/m^2 K)

TABLE II–1. LIST OF UNCERTAIN INPUT PARAMETERS FOR LARGE SCALE TEST FACILITY CALCULATIONS (cont.)

No.	Parameter	Ranges		Reference	Distribution	Explanation
		Min.	Max.			
Convergence criteria, heater power						
46	EPS	10^{-4}	10^{-2}	10^{-3}	Triangular	Convergence criterion (upper local relative error)
47	QROD0/00	0.99	1.01	1	Uniform	Correction factor for heater power (nominal: 10 MW maximum power)
48	CLIMX	0.1	1	0.2	Uniform	Correction factor for lower local absolute error of void fraction (factor 1:5 × 10^{-4})

II–1.1.3. Results

A total of 100 ATHLET calculations were performed. According to Wilks' formula, a minimum of 93 runs are required to establish two-sided tolerance limits with 95% probability and 95% confidence. Thus, at any point in time, at least 95% of the combined influence of all considered uncertainties on the calculated results lies within the presented uncertainty range, at a confidence level of at least 95%.

Of special interest is the highest calculated cladding temperature. As can be seen from Fig. II–1, the experimental measurements in the elevation showing the high temperatures (level 8) are generally inside the calculated uncertainty range. The calculated range of the second heat-up is slightly earlier than measured. The end of this heat-up is due to an early accumulator injection start due to a low range of calculated pressure in the primary coolant system.

The measured value for the first PCT is 736 K, and the calculated upper uncertainty limit is 845 K. The first PCT is not measured at level 8 (shown in Fig. II–1); it is measured at level 5 (i.e. the middle of the core). The most pronounced second heat-up, however, is measured at level 8, the upper part of the heater rods. Sensitivity measures indicate the influence of the uncertainty of input parameters on the first PCT. For example, the Spearman rank correlation coefficient is used as a sensitivity measure in Fig. II–2. The length of the bars indicates the sensitivity of the respective input parameter uncertainty on the result (here the PCT). The sensitivity measure gives the variation of the result in terms of standard deviations when the input uncertainty varies by one

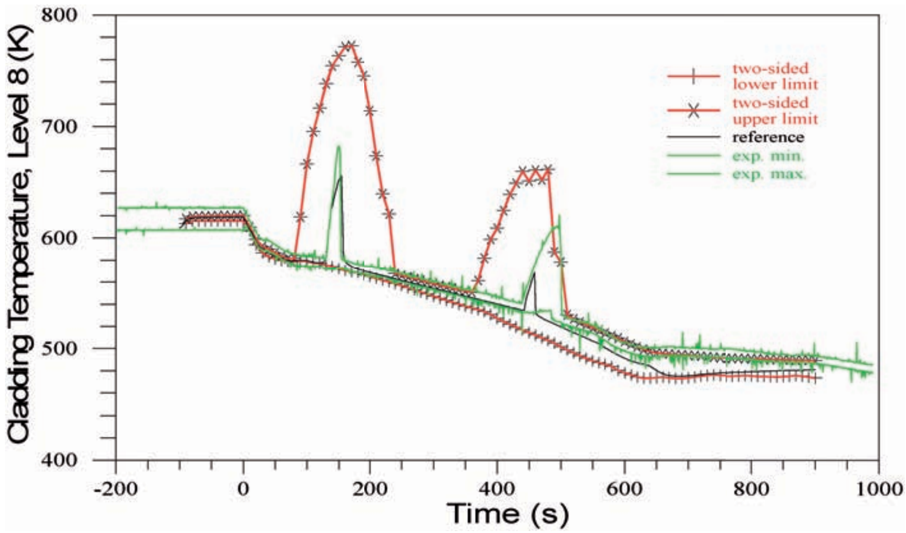

FIG. II–1. Calculated uncertainty range and BE reference calculation compared with measured minimum and maximum values of rod cladding temperature at level 8 in the LSTF-SB-CL-18 experiment.

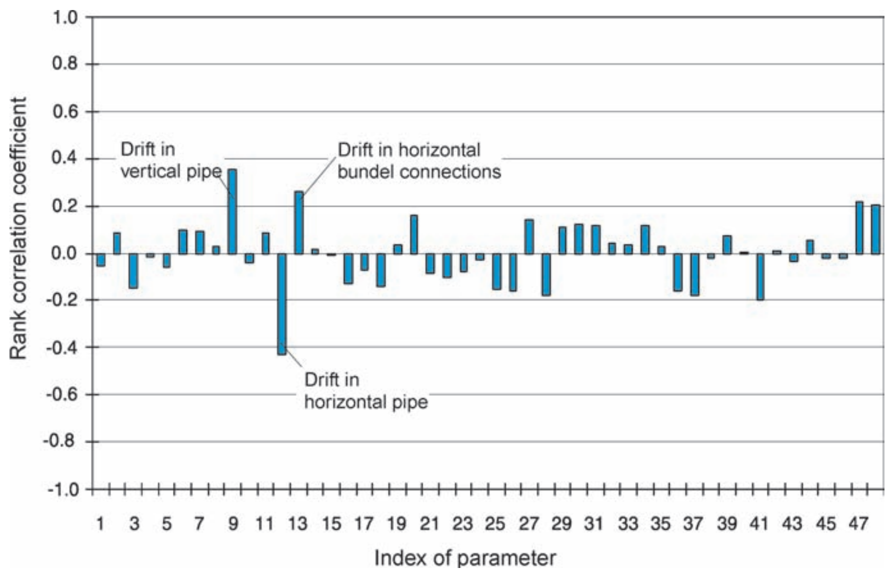

FIG. II–2. Sensitivity measures of the first PCT with respect to the selected 48 uncertain input parameters (rank correlation coefficient) for the LSTF experiment.

standard deviation (if the input uncertainties are independent). The positive values mean that the input uncertainty and the result have the same direction (i.e. an increase of input uncertainty values tends to increase the cladding temperature, and vice versa). In the case of the negative values, the input uncertainty and the result go in opposite directions (i.e. increasing the parameter values tends to decrease the cladding temperature, and vice versa).

According to these quantities, the most important three parameters are: drift in the horizontal pipe, drift in the vertical pipe and drift in the horizontal connections of the heater rod bundle. An increased drift in the horizontal bundle connections (decreased water droplet transport to the hot bundle regions) and increased drift in the vertical pipe (impedes loop seal clearance) tend to increase cladding temperature, whereas increased drift in the horizontal pipe impedes loop seal filling and results in decreased cladding temperature.

The experimental value of the second PCT is 610 K, and the calculated upper uncertainty limit is 660 K. The Spearman rank correlation coefficient in Fig. II–3 shows the top ranking of the parameters: contraction coefficient and vertical drift in the heater rod bundle. An increased contraction coefficient will lead to an earlier accumulator injection and, consequently, tends to decrease the cladding temperature. A higher drift in the bundle results in an increased cladding temperature in the upper bundle region.

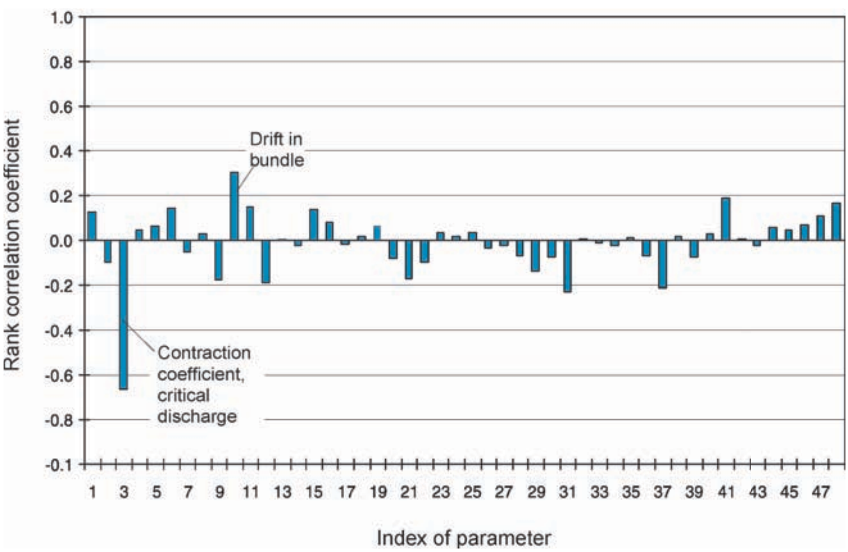

FIG. II–3. Sensitivity measures of the second PCT with respect to the selected 48 uncertain input parameters (rank correlation coefficient) for the LSTF experiment.

ACKNOWLEDGEMENT

This work performed by GRS was funded by the German Federal Ministry for Economy under Contract RS 1109.

II–1.2. Application to a German reference reactor

II–1.2.1. Description of the accident scenario

A 5% break in the cold leg of a German PWR of 1300 MW(e) is investigated. As in the LSTF experiment, a loss of off-site power at scram is assumed. The high pressure injection system is assumed to fail (this assumption is beyond design basis). All eight accumulators are available; four are connected to each of the four hot legs and four to each of the cold legs. The accumulator system is specified to initiate coolant injection into the primary system below a pressure of 2.6 MPa. After about 500 s, injection is into the hot legs only, because the cold leg accumulators will be closed. The low pressure injection system is activated at 1.06 MPa.

II–1.2.2. Uncertain parameters

All parameters identified as potentially important are included in the uncertainty analysis. For this analysis a total of 45 potentially important uncertain parameters are identified. Included are 38 model parameters, two uncertainties of bypass flow cross-sections in the reactor vessel (between the upper downcomer and the upper plenum, as well as the upper downcomer and the upper head), four uncertainties of reactor plant conditions and one uncertainty of the numerical solution procedure.

II–1.2.3. Model uncertainties

For this reactor application, 34 parameters characterize computer code model uncertainties by uncertain corrective multipliers. Four additional model uncertainties are expressed by sets of alternative model formulations (i.e. two from wall heat transfer and two from hydrodynamics (drift, pressure drop)). The quantification of model uncertainties is based on the experience gained from validation of the ATHLET code.

II–1.2.4. *Scaling effects*

Possible contributions to the uncertainty of reactor calculations may come from scaling effects. Several tests of the German UPTF and UPTF-TRAM (transient and accident management) tests on a 1:1 scale were investigated through comparisons with ATHLET code calculations or with results from small scale facilities. It turned out that no additional uncertain model parameter has to be introduced to account for scaling effects.

II–1.2.5. *Reactor plant conditions*

In order to account for uncertainties in reactor plant conditions, the uncertainties in core power (100–106%), decay heat power (DIN[8] ± 10%), fuel rod gap conductance correction factor (0.885–1.63) and temperature of the cooling water in the accumulators (30–40°C) are included. For gap conductance, a normal distribution is specified; for the other parameters a uniform distribution is specified.

Realistic initial and boundary conditions are used in the uncertainty and sensitivity analysis of the reference reactor. If the specific conditions are not exactly known, they are considered uncertain. The single failure criterion, however, is taken into account in a deterministic way; it is not treated as uncertainty. This is a superior principle of safety analysis (redundancy). The probability of system failures is part of probabilistic safety analysis, not of demonstrating the effectiveness of ECCSs. For DBAs, the cooling system effectiveness has to be proved by deterministic safety analyses with regard to the available systems. The uncertainty analysis of such deterministic calculations permits a quantitative probabilistic statement about the margin between the tolerance limits of the calculation results and the acceptance limits. In the present investigation, however, the single failure of one high pressure system and the unavailability of a second high pressure system due to preventive maintenance are exceeded by the assumption of complete failure of the high pressure injection system. High pressure system failures are the worst unavailabilities in SB LOCAs.

II–1.2.6. *Results*

The highest calculated cladding temperatures are of special interest. The PCT is calculated in the upper part of the core (level 7). Figure II–4 shows the

[8] Deutsche Industrie-Norm (German Industry Standard).

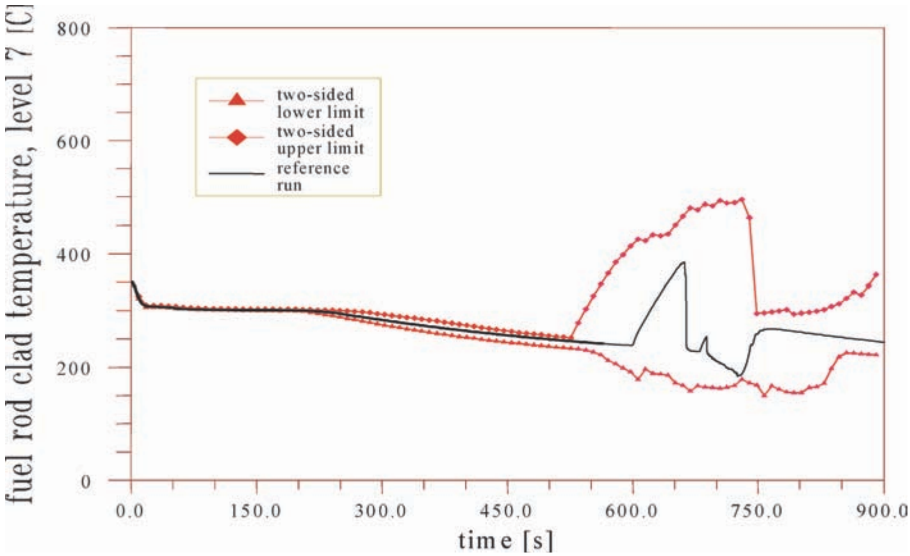

FIG. II–4. Calculated uncertainty range of rod cladding temperature and BE calculation for reference reactor.

uncertainty range for the cladding temperature of the rods at level 7. At least 95% of the combined influence of all identified uncertainties is within this range, at a confidence level of 90% (77 calculation runs were performed).

A comparison of this calculation with LSTF results shows differences at about 120–160 s. While LSTF shows a first heat-up during this time span, the reactor calculation does not show an increase in cladding temperature. This difference is mainly due to different decay power curves. The maximum power in LSTF is only 14% of the scaled power under normal operating conditions. Therefore, this highest possible power is kept for 35 s after the scram signal, and is decreased subsequently to compensate for the lower initial power. Comparisons in the LOBI experimental facility, where full scale initial power was available, in experiments with reactor typical decay heat immediately after scram, show nearly no first heat-up compared with a power curve similar to LSTF.

A heat-up is calculated at 500 s during core uncovery. The calculated upper tolerance limit of the maximum temperature at level 7 is 495°C. The maximum cladding temperature at level 7 in the reference calculation (using nominal values) is 400°C. After 750 s the uncertainty range decreases when the

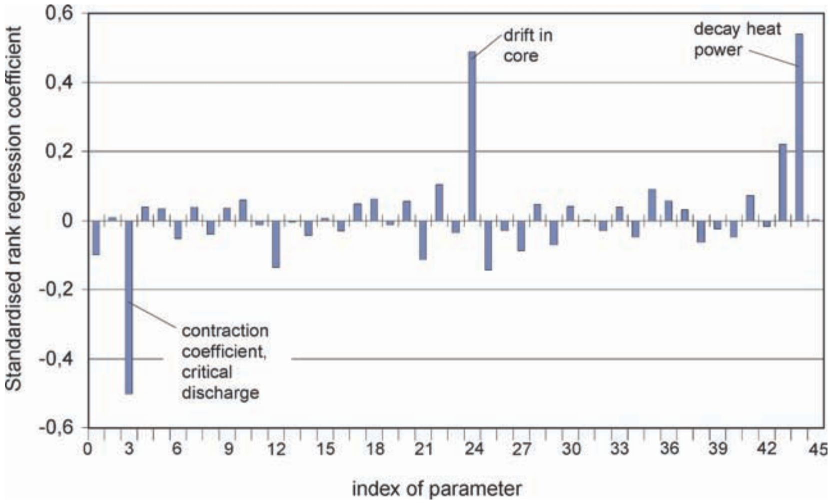

FIG. II–5. Sensitivity measures of the PCT with respect to the selected 45 uncertain input parameters (standardized rank regression coefficient) for the reference reactor.

rods are cooled due to accumulator water injection. The earliest start of accumulator injection is at 540 s.

Figure II–5 shows sensitivity measures indicating the influence of the uncertainty in input parameters on the PCT. The rank regression coefficient is shown as a sensitivity measure. Again, the length and sign of the bars indicate the sensitivity of the respective input parameter uncertainty on the result (here the PCT). According to these quantitative sensitivity measures, the main contributions to uncertainty of the PCT come from the decay heat power (parameter 44), the vertical drift model in the core (parameter 24) and the contraction coefficient of the critical discharge model (parameter 3). Increasing the contraction coefficient tends to decrease the PCT, and vice versa. An increased drift in the core and an increased decay heat power result in increased cladding temperatures at higher elevations.

The most important sensitivities with respect to cladding temperature at level 7 versus time are presented in Fig. II–6. In addition to those mentioned for the PCT, the turbulence factor for evaporation in the critical discharge model (parameter 1) is revealed to be an important contributor to uncertainty in the time between 200 and 500 s. Between 400 and 800 s the contraction coefficient at the break is a major uncertainty contributor. Increasing contraction increases the critical mass flow out of the break, increases the velocity in the upper part of the bundle, increases the cooling and, consequently, decreases the

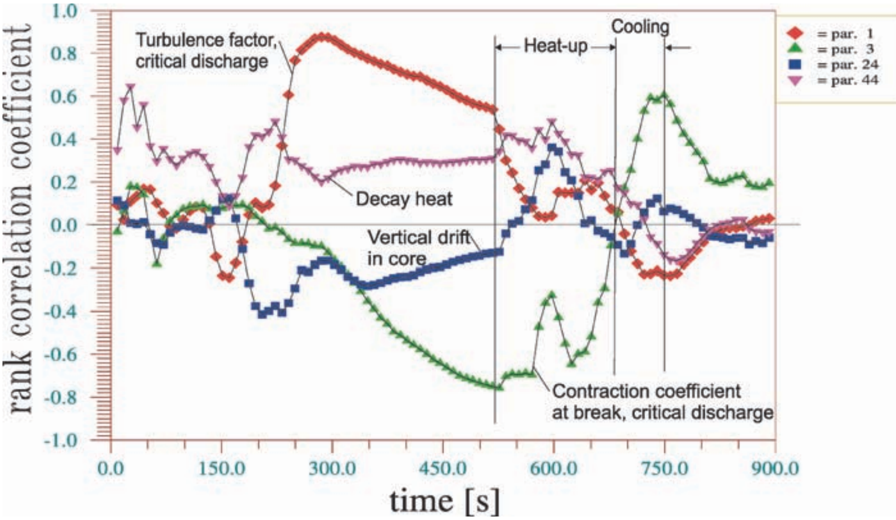

FIG. II–6. Time dependent sensitivity measures of rod cladding temperature for the reference reactor (heat-up: 530–680 s, cooling: 680–750 s).

cladding temperature (negative value). Furthermore, the pressure decrease in the primary system is faster and the actuation pressure of the accumulators is reached earlier. Thus the accumulator injection starts sooner, tending to decrease the PCT, but the sensitivity measure of the contraction coefficient becomes positive at 680 s. Increasing contraction leads to earlier accumulator injection and earlier cooling of the core, but also to an earlier increase of steam production, thus increasing the pressure in the primary system, reducing the pressure difference between the accumulator and primary system and consequently decreasing the injected emergency cooling water flow. Thus the cooling of the fuel rods is decreased earlier. Less pronounced is the contribution of vertical drift in the core. During core uncovery (530–650 s) an increasing drift between steam and water tends to increase the rod temperature. Increasing drift causes a lower water fraction in the upper core region due to lower entrainment.

ACKNOWLEDGEMENT

This work performed by GRS was funded by the German Federal Ministry for Economy under Contract RS 1109.

II–1.3. Application to the LOFT L2-5 experiment

The L2-5 experiment simulated a double ended off-set shear break of a cold leg primary coolant pipe in a commercial PWR [II–2]. The simulation was initiated from a power level of 36 MW, the maximum linear heat generation rate was 400 W/cm.

II–1.3.1. Uncertain parameters

All potentially important uncertain parameters are included in the uncertainty analysis. Included are a total of 49 uncertain input parameters, 42 model parameters, one uncertain gap width of the fuel rods between the fuel and cladding, two uncertainties of bypass flow cross-sections in the test vessel, two uncertain reactor power parameters and two uncertain convergence criteria of the code. The quantification of model uncertainties is based on the experience gained from validation of the ATHLET code.

II–1.3.2. Results

A total of 100 ATHLET calculations were performed. According to Wilks' formula, a minimum of 93 runs are required to establish two-sided tolerance limits with 95% probability and 95% confidence [II–3]. Thus at any point of time at least 95% of the combined influence of all considered uncertainties on the calculated results lies within the presented uncertainty range, at a confidence level of at least 95%.

Of special interest is the highest calculated cladding temperature. As can be seen from Fig. II–7, the experimental measurements are generally inside the calculated uncertainty range. The measured value for the blowdown PCT is 790°C. The maximum calculated upper (95%/95%) uncertainty limit is 840°C.

Sensitivity measures indicate the influence of the uncertainty in input parameters on the blowdown PCT. For example, the Spearman rank correlation coefficient is used as a sensitivity measure in Fig. II–8. The length of the bars indicates the sensitivity of the respective input parameter uncertainty on the result (here the PCT). The positive values mean that input uncertainty and results have the same direction (i.e. an increase of input uncertainty values tends to increase the cladding temperature, and vice versa). In the case of the negative values, the input uncertainty and the result go in opposite directions (i.e. increasing the parameter values tends to decrease the cladding temperature, and vice versa).

According to these quantities, the most important three parameters for the blowdown PCT uncertainty are: the gap width between the fuel and

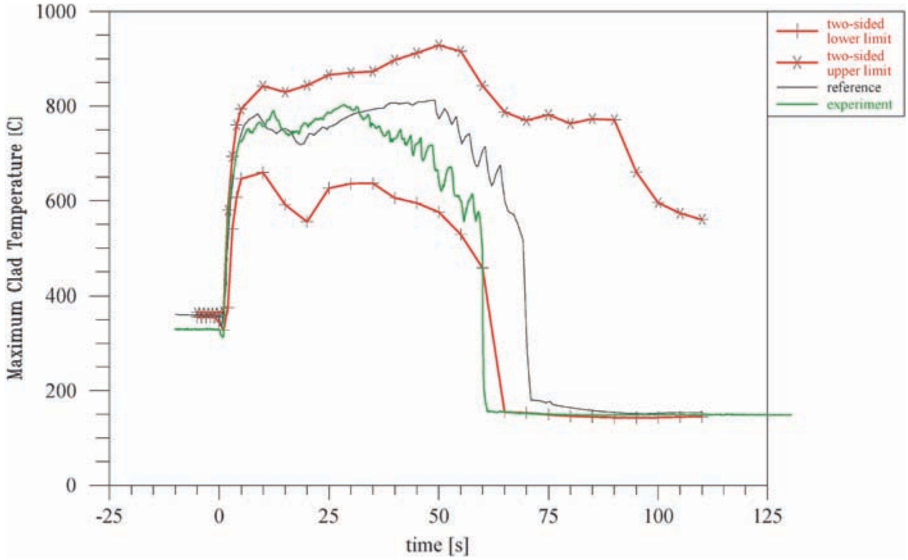

FIG. II–7. Calculated uncertainty range and BE reference calculation compared with measured minimum and maximum values of rod cladding temperature in the LOFT L2-5 experiment.

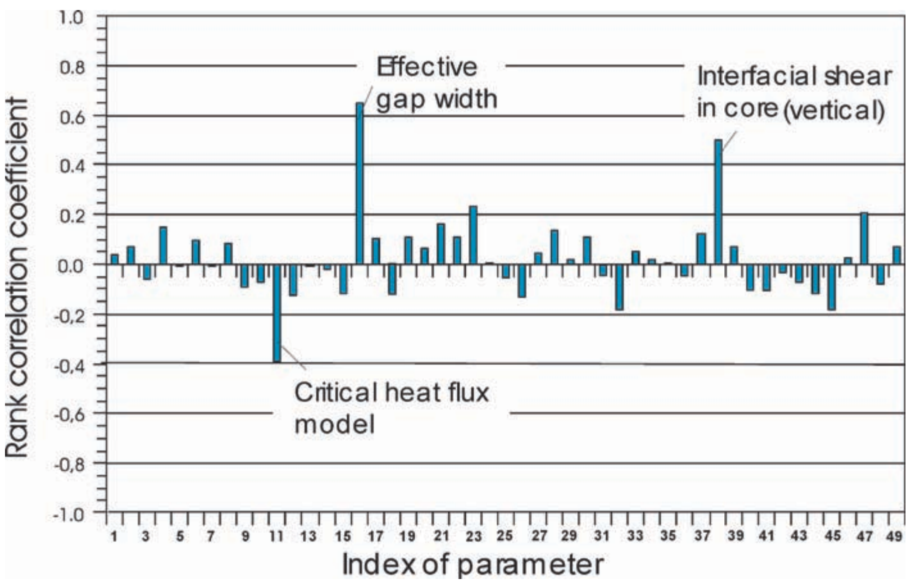

FIG. II–8. Sensitivity measures of the blowdown PCT with respect to the selected 49 uncertain input parameters (rank correlation coefficient) for the LOFT L2-5 experiment.

cladding in the fuel rod, the correction factor of interfacial shear at non-dispersed flow in the vertical bundle and the correlation for critical heat flux (CHF). An increased gap width (higher stored heat) and increased interfacial shear in the core (higher vapour fraction in the core) tend to increase the cladding temperature, whereas increased CHF results in decreased cladding temperature (later change from nucleate boiling to transition boiling).

The experimental value of the reflood PCT is 804°C and the calculated upper (95%/95%) uncertainty limit is 930°C. The Spearman rank correlation coefficient in Fig. II–9 shows the top ranking of the input parameter uncertainties for the reflood PCT uncertainty: interfacial shear at non-dispersed flow in the vertical pipe, critical velocity of change from separated flow to slug flow in the horizontal pipe and two phase multiplier in the vertical core and pipe. An increased interfacial shear in the upper plenum will lead to higher entrainment (lower water inventory) and, consequently, tends to increase the cladding temperature. A higher critical velocity in the horizontal pipe leads to a later change to slug flow, which results in a higher steam flow during separated flow conditions and, consequently, tends to decrease cladding temperature in the core bundle region. A higher two phase multiplier in the core occurs at a higher water fraction, which, in turn, reduces the cladding temperature.

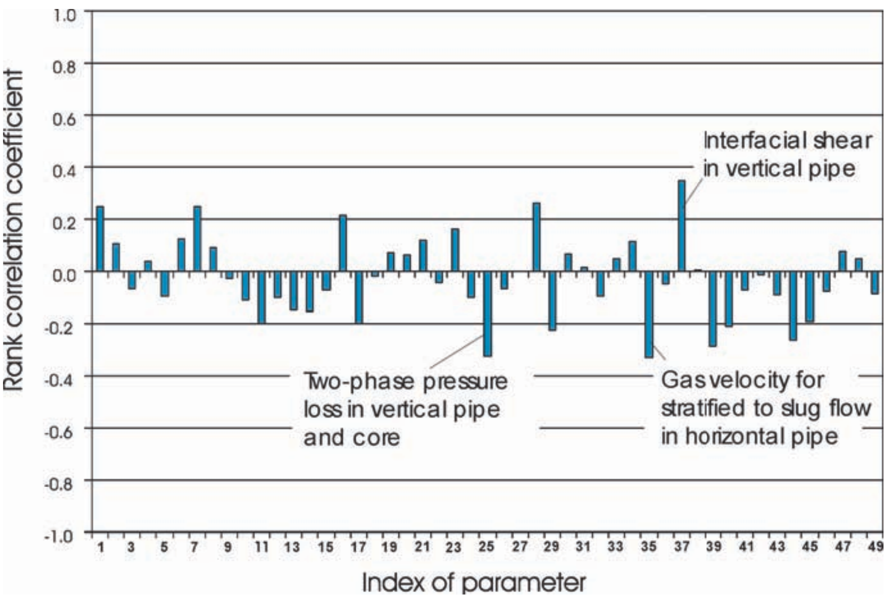

FIG. II–9. Sensitivity measures of the reflood PCT with respect to the selected 49 uncertain input parameters (rank correlation coefficient) for the LOFT L2-5 experiment.

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This work performed by GRS was funded by the German Federal Ministry for Economy under Contract RS 1109.

II–1.4. Application to a German reference reactor

II–1.4.1. Description of the accident scenario

A two times 100% guillotine break in the cold leg DBA of a German PWR of 1300 MW(e) is investigated. Loss of off-site power at scram is assumed. ECC injection is into cold and hot legs. The accumulator system is specified to initiate coolant injection into the primary system below a pressure of 2.6 MPa. High and low pressure ECC injection is available. A single failure is assumed in the broken loop check valve, and one hot leg accumulator is unavailable due to preventive maintenance. The calculations are performed using the code ATHLET Mod 1.2, cycle D [II–4].

II–1.4.2. Uncertain parameters

All potentially important uncertain parameters are included in the uncertainty analysis. Included are a total of 56 uncertain input parameters, 41 model parameters, two uncertainties of bypass flow cross-sections in the rector vessel, one uncertain temperature of accumulator water, one uncertain reactor power, one uncertain decay heat, one uncertain radial peaking factor, one uncertain hot channel peaking factor, five uncertain gap widths of the fuel rods between the fuel and cladding (for five different burnups), one uncertain heat conductivity of the fuel pellets and two uncertain convergence criteria of the code. The quantification of model uncertainties is based on the experience gained from validation of the ATHLET code.

II–1.4.3. Results

A total of 100 ATHLET calculations were performed. Figure II–10 shows, at any point of time, that at least 95% of the combined influence of all considered uncertainties on the calculated results is below the presented uncertainty limit (one-sided tolerance limit), at a confidence level of at least 95%. For comparison a ‘conservative’ calculation result is shown, applying the BE code ATHLET with default values, and conservative values for the initial and boundary conditions reactor power, decay heat, gap width of fuel rods between the fuel and cladding, fuel pellet conductivity, temperature of the accumulator water and the Baker–Just correlation for cladding oxidation (instead of the Cathcart correlation in the uncertainty analysis). All these conservative values (except the oxidation correlation) were included in the distributions of the input parameters for the uncertainty analysis. The maximum cladding temperature does not always bound the 95%/95% one-sided tolerance limits of the uncertainty analysis.

The ‘conservative’ calculation is representative of the use of BE computer codes plus conservative initial and boundary conditions. An evaluation of this kind is possible in the licensing procedure of several countries. The uncertainty of code models is not taken into account. The selection of conservative initial and boundary conditions will bound these model uncertainties. This is naturally not the case for the whole transient. An uncertainty analysis gives a better quantification, including model uncertainties. Therefore, the US Code of Federal Regulations [II–5] requires that “uncertainties in the analysis method and inputs must be identified and assessed so that the uncertainty in the calculated results can be estimated” when a BE computer code is used for the analysis.

According to the US Code of Federal Regulations, the conservative method requires that conservative models be applied in conformity with the

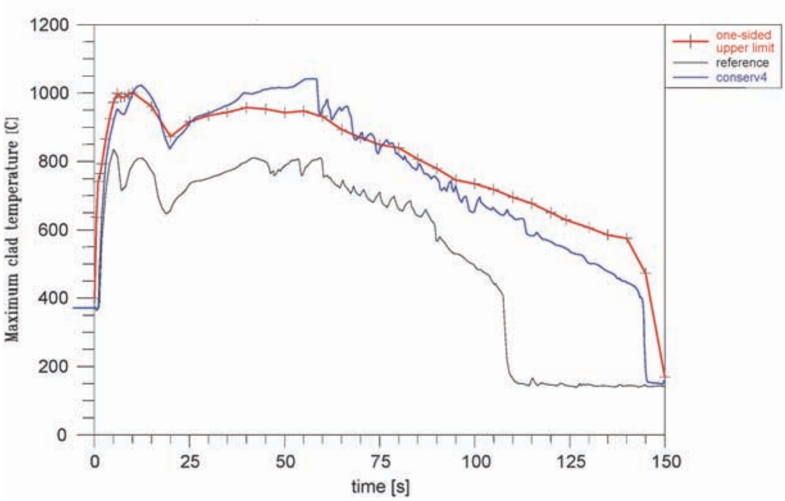

FIG. II–10. Calculated uncertainty range and BE reference calculation compared with a ‘conservative’ calculation of rod cladding temperature for a reference reactor during a postulated 2 × 100% guillotine cold leg break.

required and acceptable features of Appendix K, ECCS Evaluation Models [II–5]. This is the main reason why, in the USA, an additional margin for licensing criteria is available by changing from conservative evaluation to BE calculations plus uncertainty analysis.

ACKNOWLEDGEMENT

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II–2. CSAU

II–2.1. Introduction

CSAU was first demonstrated by the NRC, and reported in 1989. Subsequently a number of organizations used the CSAU framework to build variants of the NRC approach or to build methodologies that are only similar to the NRC approach through the CSAU framework itself.

The appeal of the CSAU framework, at least in the USA, stems from the NRC's tacit approval of the original process and hence the implied promise that BEPU methodologies based on the CSAU framework will require fewer resources to gain approval by the NRC licensing authority.

Originally the CSAU framework — and the uncertainty methodology that was first implemented based on it — was conceived to provide a means to meet the revised rule (US licensing requirements) on the acceptance of ECCSs such that more realistic estimates of plant safety margins might be made and used for the licensing of nuclear power plants.

Following the revision of the rule in 1988, the NRC sponsored two demonstrations of the CSAU process, one for an LB LOCA using TRAC [II–6] and one for an SB LOCA using RELAP5 [II–7]. Since the completion of these studies at least two nuclear fuel manufacturers have developed variations of the CSAU methodology, have obtained approval for using their methodology on their commercial reloads and have begun using them in the industry. In addition to these commercial applications, several studies have been performed to evaluate ways in which the uncertainty methodology based on the CSAU framework can be streamlined and improved.

The overall features of the CSAU framework will be discussed in this section together with some observations relevant to how the framework can be used in a streamlined fashion. The summary of the CSAU framework is followed by applications to quantify uncertainties for the LB LOCA and SB LOCA transients identified. Since the original applications of the CSAU framework gave only a single valued result with its associated uncertainty, it is important to know and consider that the CSAU framework can be modified relatively easily to obtain continuous valued results associated with their accompanying uncertainties.

II–2.1.1. Existing approaches

The original use of the CSAU framework was based on an NRC accepted means[9] of obtaining a rigorous evaluation of the important parameter uncertainties for nuclear facilities, using expert committees, numerous sensitivity calculations, response surfaces, etc. [II–6, II–10, II–11]. Subsequent applications, using the CSAU framework, have reduced, or attempted to eliminate, the involvement of the expert committees (e.g. the use of experts for only the first time scenario considerations) as well as to eliminate the lengthy process that makes use of response surfaces. There have been proposals to replace these functions, for example with expert system driven databases and Latin hypercube sampling (LHS) [II–12]; such considerations have been considered in other areas of research such that there could be a sensitivity calculation set of reduced size [II–13]. Other approaches are also feasible.

One implementation [II–11] of the CSAU framework fits calculated results by a response surface and then uses the response surface function as a surrogate for the code. The input parameters are then sampled from their uncertainty distributions and the corresponding value of the response surface function is calculated. After many such calculations, the uncertainty distribution of the response surface approximation to the code is known.

In an example problem using the response surface method, seven uncertain parameters required 207 code calculations. This large number of runs is a disadvantage of this application of the CSAU framework in its response surface form. A second disadvantage is that an appropriate response surface function must be selected (e.g. quadratic, cubic, quartic, with or without cross-product terms, or using a non-polynomial form), avoiding both underfit and overfit. While this can conceivably be automated with an expert system, it is not easy to do.

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[9] The CSAU process was applied to perform LB LOCA best estimate analyses to show conformance with the 10 CFR 50.46 requirements for Westinghouse three- and four-loop PWRs [II–8]. This application was approved by the NRC [II–9].

A simple Monte Carlo sampling could be used instead of a response surface. A virtue of this approach is that it can be automated, and it allows confidence statements on the percentiles. However, simple Monte Carlo sampling is not as efficient as other Monte Carlo methods. Consequently, an LHS method is introduced as an example simply to display the potential of the CSAU framework.

II–2.1.2. CSAU: Description of framework and summary of implementation potential

Conceptually the framework has three elements (see Fig. II–11 and Ref. [II–6]). These are described below and a summary of the original application of the CSAU framework is outlined. Thereafter some potential modifications to the original process are introduced and discussed in more detail.

Element 1. Requirements and code capabilities. The steps included in element 1 are to: (1) specify the scenario; (2) select the nuclear power plant; (3) identify and rank the processes; (4) select the 'frozen' code; (5) provide the code documentation; and (6) determine the code applicability. Element 1 was implemented by making use of the collective judgement of a panel of experts (step 3) that exercised the PIRT process to identify the key phenomena that either control the transient behaviour or exert considerable influence on the transient path.

The scenario, nuclear power plant and 'frozen' code had been selected beforehand by the NRC for the first application; these items were thus a predetermined initial condition for the first application. Consequently, steps 1, 2, 4–6 largely follow from the specified initial condition. However, the means for achieving the objectives of step 3 can potentially be modified.

This baseline application of the CSAU framework, in particular step 3, can be altered to make use of data extracted from expert committees and condensed into a form accessible to an expert system. This information could then be implemented into a computer code environment. To analyse a particular scenario, the analyst, either making use of the judgements of the panel of experts or aided by recommendations from the computer based expert system, would be able to prune the list of uncertain parameters to include only the most important ones. Using either approach, the important parameters can then be assigned an uncertainty distribution, either the default suggested by the computer or the analyst's choice. The default uncertainty distribution, if no information is available, is the uniform distribution, as used in the baseline implementation of the CSAU framework.

Element 2. Steps 7–10 constitute element 2; that is, step 7 establishes an assessment matrix, step 8 defines the nuclear power plant nodalization, step 9

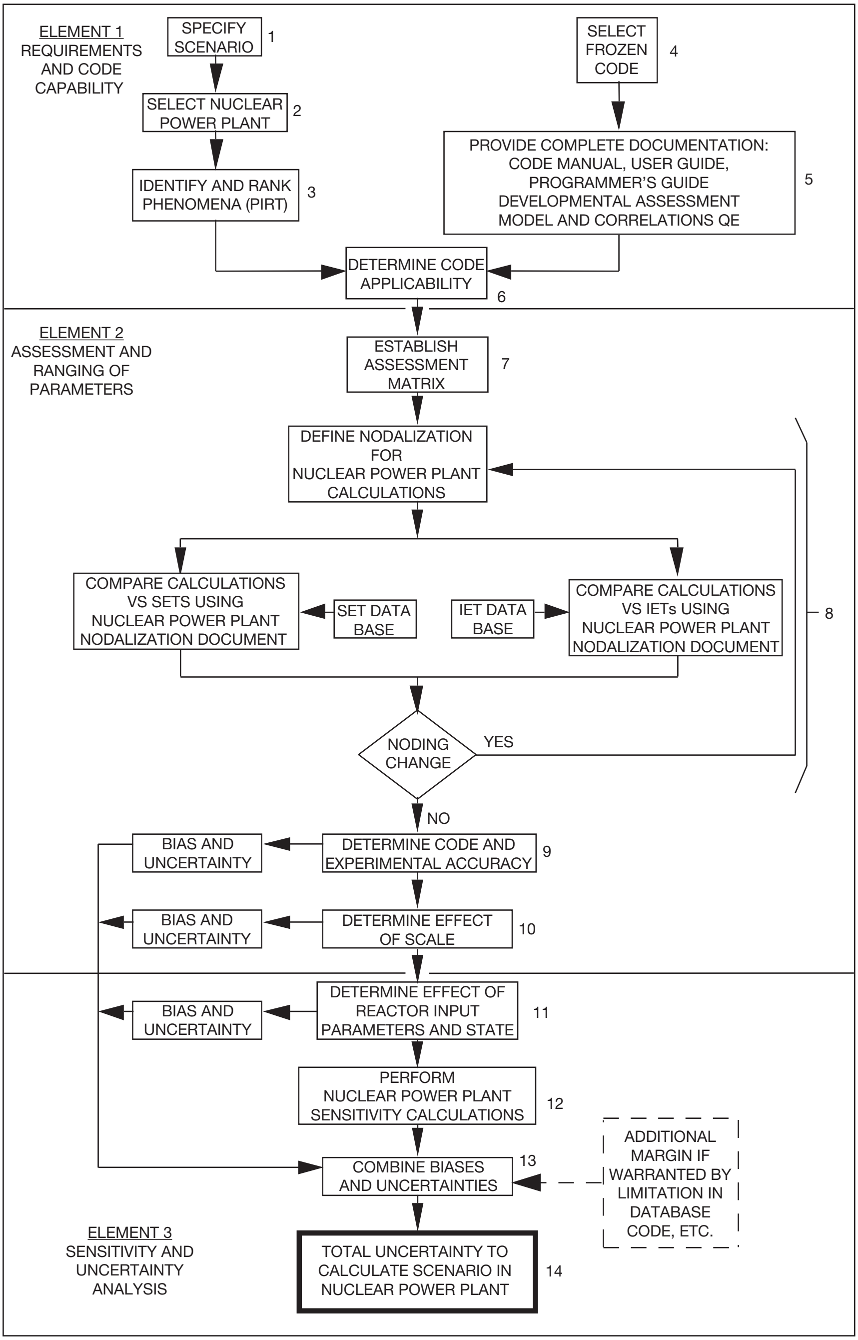

FIG. II–11. Code scaling, applicability and uncertainty framework.

determines the code and experimental accuracy and step 10 determines the effect of scaling. These steps take the uncertainty analysis to the point at which the uncertainty calculations can be performed. In short, following the determination that the code to be used is applicable to the problem slated for analysis (the final step of element 1), the experimental data sets either available or required for comparison with the code calculations are identified in an assessment matrix in preparation for creating the system analysis code model nodalization. Thereafter, the code calculations are begun to obtain calculation to data comparisons that are used to study the code calculational accuracy and then the effect of scaling. The work done in this element requires experts competent in the use of the system analysis code, in the performance of uncertainty evaluation as well as in the development of a nodalization for the nuclear power plant and the IET and SET facilities.

Element 3. The final element of the process contains steps 11–14. Step 11 determines the effect of reactor input parameters and state, step 12 performs nuclear power plant sensitivity calculations, step 13 combines biases and uncertainties and step 14 determines the total uncertainty. The ultimate outcome of this element is the calculational uncertainty for the desired performance measure, for example the PCT or the system primary inventory level. Of these four steps, steps 11 and 12 entail running a multitude of calculations using the system analysis code to study the effects of various combinations of reactor initial conditions and other boundary condition parameters. To obtain formulations that enable nuclear power plant sensitivity calculations to be performed, step 12 included the creation of the response surface, based on regression analysis via a multinomial least squares fitting process of the calculated PCT in terms of the highly ranked variables. The response surface enables the replacement of the code by a fit to the output of interest [II–11] (i.e. the PCT in this case). This process is time consuming and expensive. The baseline implementation of the CSAU framework used the response surface to simulate variable behaviour for use in predicting potential system behaviour scenarios. Alternatively, the LHS approach could be used by allowing the software to draw a random value from the distribution of each parameter, and then writing the corresponding input deck for the reactor code. An automated approach would then aim to run the code and obtain outputs of the results. Used in this way, LHS is a kind of stratified Monte Carlo sampling, which has been used for large non-linear codes such as large fault tree codes and the codes used to evaluate the Waste Isolation Pilot Plant [II–14]. The code is run once for each set of sampled parameters. The collection of outputs can be analysed in various ways. For example, a histogram of the results from the code output gives the uncertainty distribution, and the sample correlation coefficients show which input parameters are most influential in the output calculation. This process will require fewer runs than simple Monte Carlo sampling. It will also require fewer runs (half as many in the example problem discussed above) than response surface modelling, while allowing consideration of more uncertain parameters in an automated way.

It is noted that a large fraction of the process can be automated by coupling an expert system database (used in element 1) and the LHS capability (used in element 3) together with the capability to create a set of parametric calculations and an output analysis capability in an interactive environment.

II–2.1.3. Study to examine the feasibility of using Latin hypercube sampling or a similar approach

A preliminary scoping analysis considering the major elements of an LHS process[10] [II–13] was performed as an illustrative exercise [II–15], using a LOFT LB LOCA experiment as a basis. The results show that the approach:

(a) Gives the uncertainty of the desired safety criteria as a continuous function compared with an uncertainty only at the peak value using response surfaces [II–16];
(b) Is straightforward, efficient and thus should be easy to automate.

The results of the analysis (see Fig. II–12) gave the calculated PCT and the calculation uncertainty envelope as a function of time[11]. Using the methodology described in this report, similar results would be expected for the required safety criteria of advanced nuclear systems. The approach summarized herein could be used with any of the advanced thermohydraulic analysis codes, for example TRAC and RELAP5.

II–2.2. Objectives: Uncertainty methodology variants based on the CSAU framework

Streamlined uncertainty methodologies that are based on the CSAU framework, incorporate a means to replace response surfaces, use expert systems and include a means to implement automation give a BEPU process with the following features:

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[10] A simple Monte Carlo approach, not LHS, was used and the method was not automated.

[11] The original figure also showed the LOFT data inside the calculation uncertainty envelope.

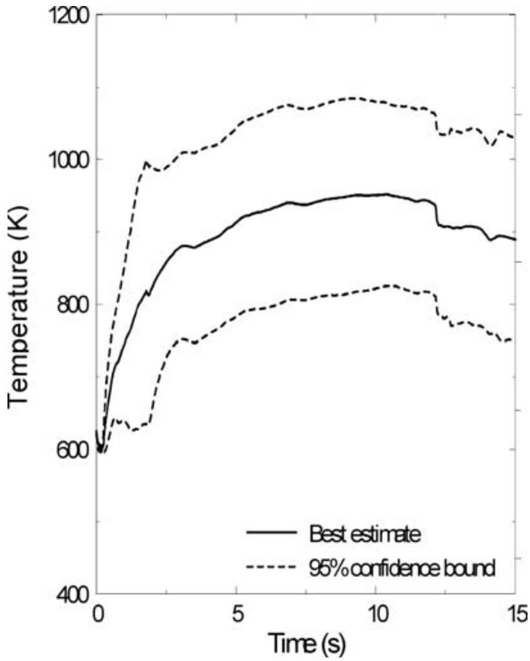

FIG. II–12. PCT with 95% confidence bounds for LB LOCA in LOFT.

(a) Easy to use:
   (i) Available information will be accessible to the user through expert systems;
   (ii) Expert systems enable the process to be automated, with the exception of boundary conditions that the user must provide.
(b) Computationally efficient.
(c) Generic in principle and applicable to new codes for systems that have not yet been designed.
(d) Contains the basis for possible extensions, such as the ability to recommend to the user potentially important parameters based on past analyses of similar problems.

II–2.3. Outline of potential uncertainty methodology variants based on CSAU

Improved variants of the original implementation of the CSAU framework [II–6, II–7, II–10, II–16] can easily be accommodated. In summary, the CSAU process consists of three elements and 14 steps (see Fig. II–11). These steps are outlined in Table II–2, together with the areas in which the methodology variants are applicable.

CSAU steps 1, 2, 4, 5 and 6. Several of the steps in the CSAU process need no modification at this juncture for nearly any variation. In particular, the need to produce a calculation that describes a specific transient will allow the user to pass quickly through steps 1 and 2 and 4–6 for most applications, since the nuclear system (and the quantity of available information on the system) will be known from the start. Regarding step 4 (selection of a frozen code), it is possible that the code chosen by the user does not have the capability to analyse the transient of interest (see the discussion in Ref. [II–17] and in Section 5.3 of this report). This problem arises when the analyst does not know the specific range of the phenomena[12] that are important during the system transient. (The phenomena of importance are identified in step 3.) Or, on a more basic level, the analyst may not know whether the code of choice has the capability to analyse the phenomenon[13]. Such questions can only be resolved by reviewing the code documentation (step 5) to determine the code applicability (step 6). These steps were performed manually in the original application [II–6], but they can be automated using expert systems.

CSAU step 3. In general, the analysis of uncertainty requires careful initial consideration of the possible uncertain parameters or calculations in the code. For each parameter or calculation, the analysts must determine a rough estimate of the uncertainty and decide which uncertainties are important enough to require modelling [II–16].

CSAU steps 7 and 8. The assessment matrices (dependent on plant and transient type [II–18], availability and structure of existing plant or system nodalizations [II–19] and comparisons between the assessment data and the subject code calculation [II–20]) can be condensed into a form to be used by the expert systems. The data given in the assessment matrix, combined with the code calculations that enabled data calculation comparisons to be made, have yielded measures of the code uncertainties for each relevant calculated treatable uncertainty parameter. The measures of the code uncertainty for the treatable uncertainty parameters will also be inserted in the expert system database. Other treatable uncertainty parameters, such as peaking factors and

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[12] For example, the critical mass flux may range from 3000 $kg{\cdot}m^{-2}{\cdot}s^{-1}$ to 10 $kg{\cdot}m^{-2}{\cdot}s^{-1}$ for the transient of interest.

[13] For example, sometimes a multidimensional analysis is required, but many advanced thermohydraulic codes can only be used to produce a 1-D analysis and/or quasi-multidimensional results.

TABLE II–2. SUMMARY OF UNCERTAINTY METHODOLOGIES BASED ON THE CSAU FRAMEWORK: VARIANTS OF ORIGINAL APPLICATION

Step No.	Objective of the CSAU step	Effect of using variant (e.g. LHS and expert systems)
1	Specify scenario: the transient to be analysed	
2	Select nuclear power plant: the plant where the transient will take place	Specified by the user
3	Identify and rank phenomena (PIRT): the phenomena of importance are either defined by an expert committee or extracted from a database	Data are inserted into the expert system database This is an ingredient of element 1 Automation of expert systems may be used to address much of the process
4	Select frozen code: the advanced thermohydraulic code (or other analysis code) is chosen	Specified by the user
5	Provide complete documentation: the documentation is provided by the code development group	Information provided to the user Decision to use the code is justified based on existing documentation
6	Determine code applicability: although decisions as to whether the code is generally applicable to plant type and transient type are made by the user, more subtle questions such as whether the code's operational envelope (defined by range of correlation applicability and sometimes phenomena type) is appropriate must be determined by user examination of the code documentation; this information can be included in the database	Currently such questions are based on documentation in step 5 This information can be put into the expert systems database
7	Establish assessment matrix: the assessment database follows from steps 1 and 3 The database exists for commercial on-line systems and some advanced systems For advanced systems with no assessment database, applicable data may be drawn from the database of similar or analogous systems	Data are inserted into the expert systems database Automation of expert systems can be performed

TABLE II–2. SUMMARY OF UNCERTAINTY METHODOLOGIES BASED ON THE CSAU FRAMEWORK: VARIANTS OF ORIGINAL APPLICATION (cont.)

Step No.	Objective of the CSAU step	Effect of using variant (e.g. LHS and expert systems)
8	Create model nodalization Compare calculations and data using standard nodalization: for commercial on-line plants and advanced systems, use existing standard nodalizations and comparisons between calculations and subscale data	
9	Code and experimental accuracy (bias and uncertainty): bias and uncertainty follow through comparison of calculations and data in step 8 Results, when known, are stored in the expert system database	Can be completed using expert systems
10	Determine the effect of scale (bias and uncertainty): scale effect in this study will be evaluated similarly to the original study Following completion of the study for one transient or plant type, the result will be stored in the expert system database	
11	Determine the effect of reactor input and state (bias and uncertainty)	
12	Perform sensitivity calculations	Based on automated output of LHS runs
13	Combine biases and uncertainties	Perform LHS runs
14	Obtain total uncertainty of highly ranked phenomena: final outcome of interest	Based on automated output of LHS runs

fuel thermal conductivity, have known uncertainties that can be input directly into the expert system database.

CSAU steps 9–11. The bias and uncertainties resulting from inaccuracies of the code calculations and experimental accuracy (step 9), the effect of scale (step 10) and the reactor input parameters or reactor state (step 11) have been evaluated for many of the treatable uncertainty parameters. Once evaluated, known biases or uncertainties can be entered into the expert system database. Examples of evaluated uncertainties and biases (including many studies that contain the necessary information to allow evaluation, but do not yet contain the specific bias or uncertainty numbers) can be found in Refs [II–17, II–18, II–20, II–21] for RELAP5.

Upon completion of steps 1–11, all information required by the LHS methodology is available:

(a) Scenario type: supplied by the user.
(b) System type: supplied by the user.
(c) Treatable uncertainty parameters: extracted from the expert system.
(d) Code applicability: inserted by the user manually; however, this step can be incorporated into the expert system.
(e) Relevant assessment matrix and data: extracted from the expert system.
(f) Consistent nodalizations of systems to be considered in the study: provided beforehand and available to the expert system.
(g) Assessment calculations: extracted from the expert system.
(h) Uncertainties for treatable uncertainty parameters: extracted from the expert system.

CSAU steps 12–14. Using the information available in the expert system, the algorithms developed during the LHS research effort will select the pertinent information and generate the appropriate code input. The LHS runs accomplish step 13. The output of these calculations will serve as the basis for the uncertainty calculations (step 14) and the sensitivity calculations (step 12).

The first task is the development of the necessary expert systems. This includes the placement of existing information in the expert systems chosen for the process.

II–2.3.1. Characteristics of expert systems compatible with the CSAU framework based uncertainty methodologies

Expert systems act in two ways to form the basis to automate the uncertainty process.

Ranking of parameters. First, expert systems are used to guide the user in gathering the appropriate data to use as boundary conditions for the analysis. For example, the expert systems should be used to identify the ‘highly ranked’ phenomena and thus the parameters that require uncertainty evaluation in CSAU step 3. Table II–3 gives a partial listing of the parameters for a four-loop Westinghouse plant LB LOCA evaluation and also a partial listing for other scenarios. Thus, through a query process, the expert system asks for the type of transient and plant to be analysed (CSAU steps 1 and 2). Input from the user indicating an LB LOCA transient for a four-loop Westinghouse plant leads immediately to the parameters that require uncertainty evaluation (see the third column of Table II–3).

TABLE II–3. EXPERT SYSTEM: CSAU STEP 3 DATA FOR TWO SCENARIOS AND PLANTS

Scenario	Plant	Uncertainty parameters
LB LOCA	Four-loop Westinghouse	Mass flow
		Gap conductance
		Peaking factor
		Fuel conductivity
		Fuel–fluid heat transfer
		Initial power
		ECC flow diversion
		Dissolved N_2
		Others
	Advanced system	Mass flow
		Vessel minimum inventory
		Others
SB LOCA	Four-loop Westinghouse	Mass flow
		Gap conductance
		Peaking factors
		Fuel–fluid heat transfer
		Initial power
		Others
	Advanced system	Mass flow
		Vessel inventory
		Others

The response surface method requires the list of uncertain parameters to be short. LHS sampling, as an example, allows a much longer list at no cost to the analyst. Thus the list for LHS sampling can be longer than that in Table II–3, expanding those called ‘others’ to include parameters that may make a moderate contribution to the overall uncertainty.

Knowledge of the parameters that are considered for uncertainty evaluation leads immediately to characterization of their uncertainty distributions. Expert systems will be used here as well.

Uncertainty distributions. When the uncertainties must be calculated, the uncertainty must be categorized. Two approaches are described here, with similar conclusions. The classification of uncertainty is accomplished by developing several classes of data sets. These include the following:

(1) Various experimental determinations of a number, c.
    (i) The difference between the experiments is only measurement error. The true value of c does not change.
    (ii) The difference between the experiments reflects actual uncontrolled variability in the value of c. In different replications of an accident scenario, c might vary to this extent.
(2) Various experimental determinations of a correlation function, such as a straight line $y = a + bx$.
    (i) The scatter of the data around the fitted line represents only measurement error. The true value of y is a deterministic function of x.
    (ii) The scatter of the data around the fitted line represents uncontrolled variability in the value of y. In different replications of an accident scenario, y would equal a function of x plus a random term.
(3) Comparisons of a fairly complicated calculated result to experimental data.
    (i) The difference between the experimental value and the calculated value represents measurement error only. This is virtually never the case.
    (ii) The difference between the experimental value and the calculated value represents random variation, reflecting terms or conditions that are not modelled in the calculation. This is the typical case.
(4) Expert opinion.

Typically, the data sets in classes 1–3 will be represented as spreadsheets. In past uncertainty analyses, experts have examined data such as those in classes 1–3, and have obtained Bayesian uncertainty distributions for the parameters or the calculations. In some cases, where relevant data cannot be obtained, experts must give an uncertainty distribution based on their general knowledge of the physical process; this is identified above as class 4.

From a somewhat different viewpoint, previous studies [II–6] have identified a number of types of uncertainty:

(a) Uncertainty in the initial conditions of the reactor at the onset of the modelled scenario. This is class 1b, modelling the uncertainty in the initial conditions based on known variability during operations. Care must be taken to account for known statistical correlation (lack of independence) in the various quantities describing the initial conditions.

(b) Uncertainty from calibrating the code to limited experimental data. This means that the empirical parameters in the code are estimates of uncertain parameters, and would have been better estimated if more data had been available. This is class 1 or 2 above.

(c) Uncertainty from simplifying reality into a computer model, and ignoring relatively insignificant features. This results in random scatter of experimental data around the code prediction, and is class 3b above.

(d) Uncertainty because directly relevant data are unavailable. They do not exist, or must be extrapolated beyond the experimental range. This is class 4 above. When expert judgement is used, the experts must be trained in ways to estimate uncertainty. In particular, they must learn to avoid overstating their degree of belief.

(e) Uncertainty because the experimental data may not be perfectly representative of the scenario to be modelled. For example, they may have come from a scaled-down facility, or from an experiment that did not perfectly mimic the hypothesized conditions. This is a combination of the above classes, and illustrates that the classes proposed above are not final. Data are used, but the uncertainty distribution must be expanded based on expert judgement. One approach is to use expert judgement to assign an uncertainty distribution to each data point, assessing what the data would be if the experimental conditions had truly mimicked the postulated scenario, and then to refit these constructed data. The uncertainty in the fit includes both the scatter in the data (the usual uncertainty) and the uncertainty in the individual data values (a non-standard uncertainty). Such analysis, accounting for uncertainty in the data values, is reflected in statistical methodology [II–22, II–23]. A second approach is to adjust the calculated result by expert judgement, and to estimate the uncertainty in the adjustment by expert judgement.

The user can modify the default uncertainty distributions. When doing so the user first will have the opportunity to modify the data set and will then be asked to identify the class of the data from a menu similar to that given above. For each class, the software will then use standard statistical tools to develop an appropriate way to obtain a Bayesian distribution for each uncertain parameter. That is, the computer code will model the truth as the estimate plus a random term. For example, if the data are assumed to be log-normally distributed, the random term will have a log Student's *t* distribution. This is an expert system, in that the software asks the user for some basic information and then follows established rules for computing the desired values. Afterwards, the user will be provided with some very simple diagnostic tools, such as residual plots, as a check on whether to accept what the computer software did. The calculated uncertainty distribution will be stored with the data set. The user will also be given the opportunity to replace the uncertainty distribution by a different one; in this case the user becomes the expert. The final uncertainty distribution is used by all Monte Carlo methods, including LHS. Examples follow.

As shown in Table II–4, to identify the uncertainty for the mass flow during an LB LOCA transient in a four-loop Westinghouse plant, both pump performance data and the break critical mass flow are required. The available assessment data for these parameters are shown in the table and consist of pump data from the Combustion Engineering [II–24] pump tests, the CREARE pump tests [II–25] and the 1/3 scale Westinghouse pump tests. Similarly, the break critical mass flow data consist of data from the UPTF [II–18] and the Marviken [II–26] test facility. Similar data sets are available for all the parameters that require uncertainty evaluation.

Comparisons between the above data and the selected code (e.g. RELAP5/MOD3) are also available, as required in CSAU step 9. For example, rigorous comparisons between the Marviken data and the RELAP5/MOD3 code have been obtained a number of times [II–27]. The results of such comparisons are either the basis for a judgement that the code models can be used to calculate the phenomena, or the basis for existing uncertainties or the basis to calculate needed parameter uncertainties.

Continuing with this example, consider the choking flow multiplier for mass flow in an LB LOCA. Rohatgi and Yu [II–28] compared TRAC calculated mass flow with Marviken experimental data, and examined the ratio C_D = (measured flow)/(predicted flow). They found that C_D varies randomly around a non-linear function $f(L/D)$, where L is the pipe length and D is the pipe diameter. They provided estimates of the three parameters of f and of the standard deviation of C_D around $f(L/D)$. This data set is of class 3b in the taxonomy suggested above. The value of C_D must be input into the code calculation, and for uncertainty calculations C_D must be sampled from its distribution. For the present proposed work, the spreadsheet will contain the data comparing RELAP calculated mass flows with the Marviken experimental data. These comparisons have been made in the past, and are on file. A fitting equation will be proposed, either of the form of Rohatgi and Yu or of a different form, whichever fits best, and an uncertainty distribution will be calculated. This will use standard non-linear-regression techniques from the statistical literature. The user will be allowed to accept this default, or to modify the data and/or fit a different type of equation.

The result will be an expression of the form:

$$C_D = f(L/D) + s(L/D)U$$

TABLE II–4. EXPERT SYSTEM: CSAU STEP 7 DATA FOR ONE SCENARIO, PLANT AND PARAMETER

Scenario	Plant	Uncertainty parameter	Components	Data sets
LB LOCA	Four-loop Westinghouse	Mass flow	Pump mass flow	Combustion Engineering pump data
				CREARE pump data
			Critical flow	1/3 scale Westinghouse pump data
				UPTF
				Marviken

where U has some uncertainty distribution such as standard normal or Student's t, and $s(L/D)$ allows the standard deviation to be non-constant. In the uncertainty analysis, U will be sampled as dictated by the LHS method. The resulting values of C_D will be used in the various runs of the code.

II–2.3.2. Using Latin hypercube sampling: An example leading to continuous valued uncertainty values

LHS uses stratified sampling for each uncertain parameter, and gives an unbiased estimator of the mean of a function. For example, suppose that the function during an accident scenario is the PCT, and suppose that the computer code correctly models the PCT as a function of the input parameters, with the uncertainties for the input parameters correctly expressed. Then LHS gives an unbiased estimator of the mean PCT, where 'mean' refers to the uncertainty distribution of the PCT. As a second example, suppose that the function equals 1 if the PCT $\leq t$, for some temperature t, and 0 if the PCT $> t$. The mean of this function can be interpreted as Prob[PCT $\leq t$], and LHS gives an unbiased estimator of this probability. Since t can be chosen at will, LHS gives an unbiased estimator of the cumulative distribution function of the PCT, and can be used to estimate percentiles, such as the 95th percentile of the PCT.

The advantage of LHS over simple Monte Carlo sampling is that, under very general conditions [II–29], the variance of the LHS estimator is smaller than that of the simple Monte Carlo estimator. Thus few samples are needed for equivalent accuracy. Unfortunately, it is not simple to state how much better LHS is in any particular example.

Ways to use LHS effectively when quantifying uncertainty for an advanced reactor require further exploration. In particular, two issues need investigation: confidence intervals around percentiles of the cumulative distribution function, such as the 95th percentile; and use of LHS with cells of unequal sizes.

Confidence intervals. Two methods have been proposed for finding confidence intervals using LHS, and two conservative methods might also be considered. All four methods will be examined here.

(a) One method is to replace a single LHS estimate based on many runs with a number of LHS estimates based on fewer runs. For example, instead of computing one estimate based on 120 runs, compute six estimates based on 20 runs each. Then use the six estimates to calculate an approximate confidence interval. This approach relies on the approximate normality of the estimator. It is appropriate when the estimates are approximately symmetrically distributed around the mean, but not otherwise. That is, it is appropriate when estimating the mean PCT, but may be inappropriate when estimating a percentile in the tail of the distribution by an extreme observed value.

(b) Owen [II–29] suggests a second method, at least in principle, for approximating the variance of the LHS estimator. This method will be investigated. However, it is not clear that it will work well in practice, in an automatable way, for estimating the probability that the PCT exceeds some particular t, where t is in the tail of the distribution.

(c) If neither of the above methods can be made to work, a crude method is to ignore the stratification in LHS, treat the LHS sample as if it were a simple random sample, and use the well known non-parametric estimator for the percentiles. Intuition strongly suggests that this approach is conservative. We would need to show that this conservatism is, in fact, the case, but this can probably be shown following the methods of Owen [II–29, II–30]. It can at least be investigated by simulation in special cases.

(d) Another approach, also presumably conservative, is to draw bootstrap samples (with respect to which Davison and Hinkley [II–31] provide an excellent comprehensive treatment) from the empirical distribution of the observed LHS values. Usual bootstrap confidence intervals are based on resampling from the empirical distribution $\hat{F}$, which is based on a random sample. We propose calculating $\hat{F}$ from the LHS sample, which should be better. The bootstrap samples from this distribution would not use stratification, so the resulting confidence intervals would presumably be wider than necessary.

Stratified LHS. Sometimes one knows the 'bad' direction in which to vary an input parameter. For example, decreasing the gap conductance increases the first reflood PCT in a non-linear but monotonic way. (This fact could possibly be established through thoughtful consideration by experts, and could definitely be established through a preliminary LHS analysis.) In this case, to refine knowledge of the upper tail of the distribution of the PCT, it is proposed that two coarse strata be defined, one including gap conductance values above the median and the other including gap conductance values below the median. These coarse strata should not be confused with the stratification inherent in LHS methods. Any set of values for the p parameters corresponds to a point in p dimensional space, and the two strata defined here partition p dimensional space into two regions. A small, fixed number of values would be sampled from the first stratum, by LHS or a simple Monte Carlo method, and a relatively large fixed number of values would be sampled from the second stratum. The second stratum would either have a single large LHS sample or several smaller LHS samples that could be used to yield confidence intervals. Regardless of the details, this would focus the computing effort on the upper tail of the PCT without wasting time on the relatively uninteresting lower tail. The cumulative distribution function would be estimated in a coarse way on the left (small PCTs) and in a finer way on the right (large PCTs).

Further studies have investigated the properties of stratified LHS, comparing it with other Monte Carlo methods that have the same goal, such as LHS with unequal cell sizes [II–32], or with biased Monte Carlo sampling [II–33, II–34]. However, stratified LHS appears superior because the resulting empirical distribution function automatically has a range from 0 to 1, with no need for an empirical renormalization.

II–2.4. Automation of the process

A driver program will allow the user to carry out the analysis in as painless a fashion as possible. Figure II–13 shows the proposed flow of the program in summary form.

User software interfaces that are similar to those already used in the industry will be included [II–10, II–34]. The primary software requirements are the following:

(a) Allow the user to choose the parameters whose uncertainties are to be modelled (with the software providing a recommended ranking based on earlier expert analysis).
(b) Allow the user to choose the number of LHS runs, and to identify any uncertain parameters that should be analysed with unequal cell sizes.

Box 1: Display list of possible uncertain parameters with default importance rankings. Ask user which uncertainties should be modelled. (Expert system assistance with CSAU step 3.)

Assist user in specifying number of LHS runs and in identifying any parameters that should be sampled from one tail more than from the other.

Ask user to select any uncertainty distributions to be modified. For these distributions, allow user to modify the data, resulting in modified uncertainty distributions, or to modify the distributions directly. (Expert system assistance with CSAU steps 7 and 9–11.) Generate LHS samples for the uncertain parameters.

↓

Box 2: Write set of virtual input decks for the code, one deck for each LHS run, so that each uncertain parameter is sampled across its range.

↓

Box 3: Run the code with each input deck, possibly in parallel.

↓

Box 4: Collect the results of interest, such as peak cladding temperature or inventory, from each code run.

↓

Box 5: Display the uncertainty analysis results. These include:

- A cumulative distribution of the output of interest, such as PCT or inventory;
- Summary statistics for this distribution (moments and percentiles);
- Graphical plots of relevant time dependent quantities, such as PCT, with uncertainty bands;
- Graphical plots and numerical measures showing which inputs are most highly correlated with the output.

FIG. II–13. Overview of the program flow.

(c) Allow the user easy access to any data set, if desired.
(d) Allow the user easy access to changing the way the data set is analysed, if desired.
(e) Allow an automated way to link the uncertainties, perform LHS runs and collect the results.

This will require some modification of the reactor code, such as RELAP5, if any of the uncertain parameters are currently hardwired into the code. In each LHS run, the hardwired value must be replaced by a random value chosen by the driver program. In Fig. II–13, the value is read from a virtual input deck instead of being hardwired within the code.

In Fig. II–13, boxes 1, 3 and 5 are relatively generic. Boxes 2 and 4 require specific coding to match the input and output of the particular code being run.

II–2.5. Significance and benefits

The use of the CIAU method not only facilitates the evaluation of uncertainty but also has a number of other beneficial effects. These may be summed up as follows:

(a) Most important, an automated uncertainty methodology is a natural ingredient of a quicker, more efficient licensing procedure. This will enable advanced facilities to gain their operating licence more rapidly and to commence operation in a shorter time. This would reduce one economic barrier to the development of advanced nuclear power systems.
(b) Through the combination of uncertainty quantification with BE calculation, this methodology will encourage analysts to think about uncertainties when writing future analysis codes. The programming effort will be more efficient if the uncertainty considerations are not added to the code by later patches and add-ons. Also, analysts and programmers will be encouraged from the start to avoid the use of quantities that cannot be estimated well.
(c) The approach is generic. Therefore, the method (although not the details) will be useful for a wide variety of complex codes, not only models of advanced reactors. Spin-off benefits could apply to models of waste disposal facilities, space technology or other complex structures that require risk assessment.
(d) This expert system could form the basis for possible future extensions in the system's expertness, such as the ability to recommend to the user potentially important parameters based on past analyses of similar problems.

II–2.6. Summary: Original uncertainty analyses performed within the CSAU framework

The above sections summarize the CSAU framework and discuss and provide an example of a means of using the CSAU framework to produce uncertainty results that are continuous valued functions. It is important to bear such approaches in mind, since the original uncertainty calculations performed within the CSAU framework are now outdated. Therefore, although a summary of these is useful, their results and procedures should be viewed in their historical context as the first applications of this kind of process. At the end of this annex is provided a summary of the LB LOCA and SB LOCA analyses performed in the 1990s.

II–3. APPLICATION OF THE UMAE TO THE UMS (LSTF SB-CL-18 EXPERIMENT)

II–3.1. Test description

The LSTF SB-CL-18 test has already been described in Section II–1.1; it will therefore not be described again here.

II–3.2. Logical steps for the application of the UMAE

All uncertain parameters are part of the uncertainty study, as testified in Ref. [II–10]. The logical steps for the application of the method, as well as the detailed set of conditions for obtaining meaningful results, are listed in Ref. [II–35]. The following outline of the steps followed and of the related conditions gives some indication of how the method should be applied:

(a) SB LOCA experiments performed in the BETHSY, LOBI and SPES facilities were selected as the basis for obtaining the accuracy of the code nodalization. However, the number of experiments selected for the extrapolation of the accuracy differed between the RELAP5 code and the CATHARE code.
(b) The LSTF nodalization (reference system for this study) was developed based on criteria similar to those applied for the development of nodalizations for the ITFs referred to above.
(c) It was demonstrated that the adopted code is capable of calculating each phenomenon expected to occur during the SB LOCA under consideration (qualitative accuracy evaluation) and that its capabilities in doing so

are state of the art capabilities (quantitative accuracy evaluation). The quantitative accuracy evaluation was carried out using the FFT based method (FFTBM) [II–36].

(d) The nodalization of the LSTF was qualified at the steady state level and at the on-transient level (Kv scaled calculation) by following specifically designed procedures.

(e) The typical tolerance limits of 95% probability with 95% confidence are 'embedded' in the process of accuracy extrapolation.

II–3.3. Results and conclusion from the application

The RELAP5 and CATHARE codes were adopted to perform a reference calculation and to obtain uncertainty bands. Continuous uncertainty bands were obtained for primary system pressure and mass inventory as well as for rod surface temperature at an assigned elevation in the core simulator.

The main results are shown in Fig. II–14, and are related to the uncertainty bands predicted for rod surface temperature at the prescribed core level. For the sake of completeness, results from all methodologies applied in the UMS are given in the same figure. Related to the UMAE (second and third diagram in the first row of the figure, related to RELAP5 and CATHARE applications, respectively), the following comments apply:

(a) For both RELAP5 and CATHARE, the uncertainty bands bound the experimental data, demonstrating a successful application of the method.

(b) UMAE results are substantially similar to those obtained by the GRS method (see also Section II–1), while substantial differences appear in comparison with results obtained by the ENUSA and AEA methods. These latter differences can be explained considering the different maturity level of the various methods and the amount of resources available for their application. The study of bifurcations [II–37] as well as the outcome of a post-UMS activity [II–38] shed further light on these differences.

(c) Differences between RELAP5 and CATHARE applications can be explained through the different number of experiments at the basis of the extrapolation of accuracy and the different capabilities of each code in predicting SB LOCA scenarios.

(d) The error in predicting the PCT is of the order of 100 K for both UMAE applications. Errors in predicting the timings when temperature excursions occur can be evaluated as being of the order of 20% of the transient timing (i.e. two minutes' error in predicting the time of occurrence of a phenomenon measured at 600 s).

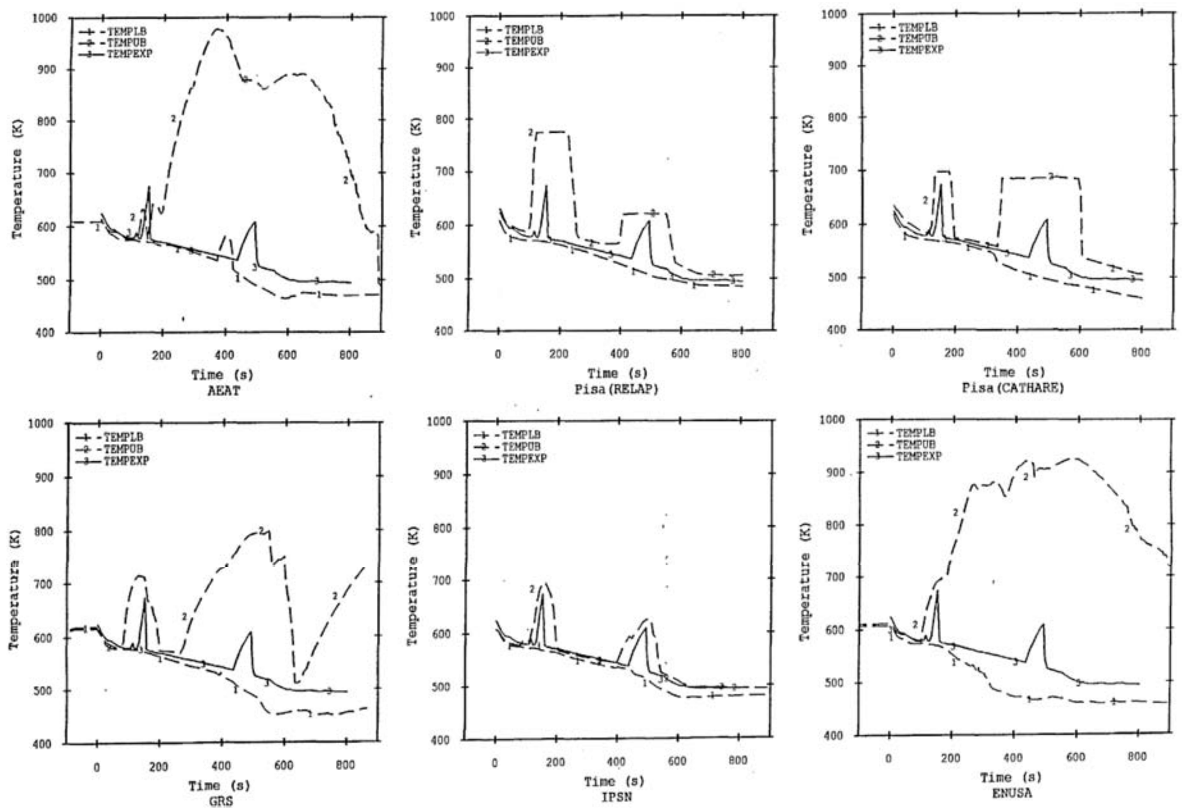

FIG. II–14. Calculated uncertainty range compared with measured values of rod cladding temperature at level 8 of the core in the LSTF-SB-CL-18 experiment, as predicted in the UMS application of various uncertainty methods. (Figure taken from Ref. [II–10].)

(e) The application of the UMAE, consistently with the results of the application of the other uncertainty methods, shows that uncertainty is not an increasing function of time and that a ‘physical’ error compensation occurs in the application of system codes to the prediction of complex scenarios.

It should be noted that almost all uncertainty methods are based upon the principle of ‘propagation of code input uncertainties’, while the UMAE method follows the principle of ‘propagation of code output error’. Both principles have associated advantages and drawbacks. The main drawbacks of the first category are the need to select a reasonable number of variables and to associate ranges of variations and, possibly, distribution functions for each of these. In addition, the uncertainty propagation occurs throughout the code itself, which, by definition, is an ‘imperfect’ tool (which is the reason why uncertainty evaluation is needed). The main drawbacks of the second category are the lack of a formal analytical procedure to obtain uncertainties and the need to have available relevant experimental data. In addition, the sources of error cannot be identified as output from the application of the methods. In the second category of methods, engineering judgement can be avoided in the phase of application of the method.

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Annex III

SUPPORTING METHODS

As already mentioned, methods are described hereafter that do not constitute self-standing approaches for uncertainty evaluation, but can be used as tools for the evaluation of uncertainty.

III–1. PHENOMENA IDENTIFICATION AND RANKING TABLES (PIRTS)

Some methods, such as CSAU, focus only on phenomena and processes that are important to the particular scenario and plant design [III–1]. The reason for doing so is to reduce all potential uncertainties to a manageable set and thus to reduce the number of calculation runs. An alternative would be to use Wilks' formula, as proposed by GRS (see Section 3.2.1.2). According to this formula, the number of calculations to be performed is not dependent on the number of input uncertain parameters but on the tolerance limits, probability content and confidence limit.

If Wilks' formula is not used, it is necessary to reduce the number of input parameters because the number of calculations would be dependent on the number p of uncertain parameters. If one were to perform a sequential variation of parameter values combining all identified uncertain parameter values, the number of calculations to be performed would increase with the power of number of parameters (see Section 5.6.5.1). This would mean that one should reduce the number of uncertain parameters significantly to reduce the number of calculation runs. Another way is to use response surfaces fitted to a small number of actual code calculation results, to substitute the BE computer code and to perform cheaper calculations.

During a phenomena identification and ranking process, physical processes are first identified (together with relevant plant components) and then ranked to establish the PIRT appropriate to the particular scenario and plant design. The identification and ranking should be justified and documented. The rationale is that plant behaviour is not equally influenced by all processes and phenomena that occur during a transient. The effort reduces all candidate parameters to a manageable set by identifying and ranking the phenomena with respect to their influence on the primary safety criteria.

III–2. THE CIRCE METHOD USED FOR BASIC UNCERTAINTIES

CIRCE stands for ‘Calcul des incertitudes relatives aux corrélations élémentaires’ (calculation of uncertainties of the elementary constitutive relationships). The CIRCE method [III–2] deals with the general problem of uncertainty analysis of BE codes and more precisely with the question of the determination of the uncertainties due to the empiricism of the closure laws (correlations). Generally, the only proposed approach is expert judgement. With CIRCE, the Comissariat à l’énergie atomique proposes a statistical method for data analysis used for the CATHARE code.

III–2.1. The problem to be solved and the solution chosen by CIRCE

The database used by CIRCE is the set of experiments considered for the qualification of CATHARE. The results of these experiments are analysed via a statistical process in order to determine the uncertainties of the closure laws. The difficulty of the problem is that, generally, the closure laws are not measurable in the experiments devoted to their study. The experimenter knows only physical quantities that are sensitive to the studied correlations, for example wall temperatures for the closure law describing the heat exchanges between wall and fluid via a film boiling process. These physical quantities are called responses.

By analysing the code–experiment differences expressed in terms of the responses ($R_{exp} - R_{code}$), and possibly the experimental uncertainties (δR_{exp}), CIRCE calculates the mean value and the standard deviation of the ε parameters associated with the studied correlations. CIRCE makes it possible to consider several closure laws simultaneously. Calculation of the statistical features of the ε parameters (mean value and standard deviation) from the $R_{exp} - R_{code}$ differences is possible without many CATHARE sensitivity calculations on account of the adjoint sensitivity method as a tool incorporated into CATHARE. This method calculates the exact derivatives of a response with respect to different parameters, that is to say $\partial R_{code}/\partial\varepsilon$ (e.g. the derivatives $\partial t_{wall}/\partial\varepsilon$ with ε associated with the film boiling exchange coefficient). This powerful tool works as a post-processing module of CATHARE and has a low CPU requirement, even for a large number of parameters. These derivatives make it possible to write a linear dependency between the parameters and the responses:

$$R_{exp} - R_{code} = \frac{\partial R_{code}}{\partial \varepsilon}\varepsilon$$

To sum up, CIRCE performs a statistical analysis on the basis of:

(a) The R_{code} code responses, obtained by standard CATHARE calculations of the considered experiment.
(b) The $\partial R_{code}/\partial \varepsilon$ derivatives, calculated with the adjoint sensitivity method post-processing module.
(c) The R_{exp} experimental responses, found in the test reports.

These quantities constitute the input data of CIRCE. Generally, only from one up to three correlations are considered together. To be reasonably precise, CIRCE needs a large number of responses, typically several tens. For one experiment, as many tests as possible must be considered, and for each test many responses.

III–2.2. Features of and possible improvements to CIRCE

The algorithm used by CIRCE is the E-M algorithm, well known in statistics. It is based on the principle of maximum likelihood, Bayes' theorem [III–3] and, as stated above, a linear approach: the model chosen for the dependence between the parameters and the responses is a linear one. Consequently, results obtained with CIRCE are valid only if the biases and the standard deviations are low. Unfortunately this is not always the case, especially with respect to the biases. This strong hypothesis of linearity can be rendered less restrictive with 'iterative CIRCE', which uses a Gauss–Newton approach for calculation of the biases. Iterative CIRCE solves the problem of non-linearities with respect to the biases. This means that the results provided by iterative CIRCE are valid even for high biases, but the calculated standard deviations should be small. In the case of high standard deviations, a new approach — which is at present under development — must be used.

III–2.3. Summary

In France, an important work programme on the systematic determination of the uncertainties of the closure laws with CIRCE was planned for Revision 6/Version v1.5 of the CATHARE2 code; the programme was to take seven person-years and to end in 2003. The work started at the end of 1999 with the analysis of the VERTICAL CANON and PERICLES boil-off and Winfrith experiments. A programme of this kind will make CIRCE the only code to be released with the uncertainties of its correlations determined by means other than expert judgement.

III–3. FAST FOURIER TRANSFORM BASED METHOD

One step common to all uncertainty methods is the use of experimental and plant data for nodalization development and qualification. When thermohydraulic computer codes are used for simulation, the questions raised are: How should improvements be added to the input model? How much simplification can be introduced? How does one conduct an objective comparison? The FFTBM assists in answering these questions. The method is easy to understand, convenient to use, user independent and clearly indicates when improvements to the simulation are necessary.

The FFTBM shows measurement–prediction discrepancies in the frequency domain, as pointed out by Ambrosini et al. [III–4]. For the calculation of these discrepancies, the experimental signal $F_{\exp}(t)$ and the error function are needed. The error function in the time domain $\Delta F(t)$ is defined as:

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

where $F_{\mathrm{calc}}(t)$ is the code predicted signal.

The accuracy quantification of a code calculation for an individual parameter is based on the amplitudes of the discrete experimental ($\tilde{F}_{\exp}(f_n)$) and of the error signal ($\tilde{\Delta}F(f_n)$) obtained by FFT at frequencies $f_n = n/T_d$, where (n = 0, 1,..., 2^m), m = (9, 10, 11, 12) and T_d is the time duration of the analysed transient. These spectra of amplitudes, together with frequencies, are used to calculate the average amplitude (AA) and weighted frequency (WF) that characterize the accuracy of the calculation. For each parameter they are defined as follows:

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

The most significant information is provided by AA, the relative magnitude of the discrepancy resulting from the comparison between the calculation and the corresponding experimental parameter time history. For a thermohydraulic transient, better accuracy is generally represented by low AA values at high WF values, as shown in Ref. [III–5].

The overall picture of the accuracy for the given code calculation is obtained by defining average performance indices, total average amplitude and total WF:

$$\mathrm{AA}_{\mathrm{tot}} = \sum_{i=1}^{N_{\mathrm{var}}} (\mathrm{AA})_i \left(w_f\right)_i \quad \mathrm{WF}_{\mathrm{tot}} = \sum_{i=1}^{N_{\mathrm{var}}} (\mathrm{WF})_i \left(w_f\right)_i \quad \text{with} \sum_{i=1}^{N_{\mathrm{var}}} \left(w_f\right)_i = 1 \tag{III–3}$$

where N_{var} is the number of the parameters analysed, and $(\mathrm{AA})_i$, $(\mathrm{WF})_i$ and $(w_f)_i$ are the average amplitude, WF and weighting factors for the ith analysed parameter, respectively.

Each $(w_f)_i$ accounts for experimental accuracy, and the safety relevance of particular parameters and its relevance with respect to pressure are specified in Ref. [III–5]. This introduces a degree of engineering judgement in the development of the method but not in its application, which has been fixed by a proper and unique definition of the weighting factors. The weights must remain unchanged during each comparison between code results and experimental data related to a single class of transient. Finally, based on several calculations, the acceptability factor K for total average amplitude as well as for AA was set to 0.4, except for primary pressure, where AA was set to 0.1.

The method has been applied to various international standard problems, standard problem exercises and other simulations of experimental data. The results show that the FFTBM is an appropriate mathematical tool for the quantitative assessment of thermohydraulic code predictions of LWR transients [III–6].

III–4. OPTIMAL STATISTICAL ESTIMATOR

The response surface can be used to solve numerous problems related to nuclear safety when thousands of complex computer code runs are needed to reach a conclusion. The optimal statistical estimator (OSE) is used to generate a response surface of complex and non-linear phenomena for single valued and continuous valued parameters. The original OSE [III–7], applicable to 1-D space, was adapted for the multidimensional space needed in the nuclear thermohydraulic field.

The response surface is predicted from the calculated or measured values. It is expressed as a linear combination of code calculated output values and coefficients representing the similarity between the code and given input data. In the case of uncertainty evaluation, this linear combination is used to replace the code calculated value when the Monte Carlo method is used to generate an approximate distribution that characterizes uncertainty in a certain parameter. The OSE H_0 and the coefficients C_n are defined as follows:

$$\hat{H}_0(G)=\sum_{n=1}^{N}C_nH_n,\quad C_n\equiv\frac{\delta_a\left(G-G_n\right)}{\sum_{n=1}^{N}\delta_a\left(G-G_n\right)} \tag{III–4}$$

where $G = (x_1, x_2,\ldots, x_M)$ is the given input data vector, $G_n = (x_{n1}, x_{n2},\ldots, x_{nM})$ and $H_n = (x_{n(M+1)}, x_{n(M+2)},\ldots, x_{nI})$ are input and output data vectors for the nth calculation, respectively, M is the number of input parameters, I is the number of output parameters and N is the number of calculated or measured values.

The approximation of the δ function is the Gaussian function:

$$\delta_a\left(G-G_n\right)=\left(\prod_{i=1}^{M}\frac{1}{\sqrt{2\pi}\sigma_i}\right)\exp\left(-\frac{1}{2}\sum_{i=1}^{M}\left(\frac{x_i-x_{ni}}{\sigma_i}\right)^2\right) \tag{III–5}$$

where σ_i is the width of the Gaussian curve selected by the user. The contribution of each data point to the final output parameter estimation can be adjusted by this function, as shown in Ref. [III–8].

To produce output results, the values of the input parameters $(x_1, x_2,\ldots, x_M)$ are randomly sampled (or input by the user) each time, and then the corresponding unknown output values are predicted by OSE using Eqs (III–4) and (III–5). Each time, new coefficients C_n are calculated, while the values of H_n are calculated points obtained by the computer code (e.g. PCT and minimum level in the core).

The major advantage of OSE with respect to regression analysis lies in its ability to predict very complex and highly non-linear functions. Also, the algorithm for OSE is suitable for computer automation, while for regression analysis statistical packages are used. The findings of the study described in Ref. [III–9] suggest that OSE can be used for response surface generation of any safety or system parameter (single valued or continuous valued) in thermohydraulic safety analyses with uncertainty evaluation.

III–5. GNOSTICAL CHARACTERISTICS METHOD

Prediction of the real thermohydraulic system behaviour is an estimate, the deviations from reality of which express the uncertainty of the prediction and thus reflect the quality of the estimate. Uncertain processes, uncertain model parameters and other unknown sources and effects are transformed into the prediction uncertainty. An estimated prediction is the result of computations performed with a selected thermohydraulic code (qualified selection) that has the capability to solve a set of problems at the state of the art level. The uncertainty of the prediction provides an indication of the computer code's suitability for the set task and the selection of appropriate models and their parameters. The prediction uncertainty analysis consists of searching for and assessing the uncertainty sources, with the aim of subsequently improving the mathematical modelling.

The uncertainty of quantities or parameters can be described by their probability distribution. The statistical methods of uncertainty analysis are thus based on the theory of probability and mathematical statistics. The study described in Ref. [III–10] outlines some of the statistical methods.

Uncertainty analysis of thermohydraulic computations is a logical follow-up to the stage devoted to the estimation of uncertainties and their quantification. Wickett et al. prepared a comparison of the methodologies (the UMS [III–11]) for the OECD/NEA–CSNI Task Group on Thermal-hydraulic Applications that showed the uncertainty bands for selected output variables of thermohydraulic computer codes to be determined.

The method, based on the Fourier discrete transformation, which is applied to evaluate the prediction accuracy, obtains the resulting criterial values from the transformed characteristics in the frequency domain. This method was applied to the evaluation of the IAEA-SPE4 test problem using four computer codes (ATHLET, CATHARE, MELCOR and RELAP5); the input data originated from the Hungarian PMK-2 facility, where an SB LOCA in the cold leg of a WWER-440 had been modelled [III–6].

The gnostical characteristics method is based on the principles of mathematical statistics; the alternative to this approach is gnostical theory, as proposed in Refs [III–12–III–14]. This is applied to the data treatment when there is a lack of data, when data are invalidated as a result of a strong uncertainty and when the mathematical–statistical model of data and failures is not known, as shown in Ref. [III–15]. Gnostical theory is not based on statistical assumptions. The theory of small data samples is obtained from the uncertainty model of the individual data and from the composition law that determines how the uncertainties of individual data are composed.

In Ref. [III–15] the application of gnostical theory to three different data samples is described:

(a) The medium amplitude samples $(AA)_{tot}$ from the test problem IAEA-SPE4;
(b) The point value prediction (PCT);
(c) The transient characteristics from the comparative methodological UMS.

The medium amplitude samples are composed of a small number of data for which the results of gnostical and statistical treatment are compared. The point prediction sample, together with the interval uncertainty estimate and experimental value, creates a data sample with two values with the reference value to which the limiting values of the interval estimate are related, and their position outside the range is evaluated. A limiting number of values in data samples of a transient uncertainty range and their implementation are applicable for evaluation of the local and global properties of the predicted transient characteristics.

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Annex IV

CODE WITH THE CAPABILITY OF INTERNAL ASSESSMENT OF UNCERTAINTY: THE CIAU METHOD (UNIVERSITY OF PISA)

IV–1. INTRODUCTION

The internal assessment of uncertainty is a desirable capability for thermohydraulic system codes to possess, as already discussed in Section 6.2. Internal assessment of uncertainty is the possibility to obtain suitable uncertainty bands each time a nuclear plant transient scenario is calculated. A methodology suitable for introducing such a capability into a system code is discussed here. The CIAU, as developed by the University of Pisa, was based on the UMAE; however, other uncertainty methodologies can also be used for the CIAU [IV–1].

The idea of the CIAU is to identify and characterize standard plant states and the association of uncertainty to each state. One hypercube and one time interval identify the plant state. ‘Quantity’ and ‘time’ uncertainties are combined for each plant state. The NRC RELAP5/MOD3.2 system code coupled with the UMAE constitutes the CIAU.

IV–2. NUCLEAR POWER PLANT STATE APPROACH

The usual characterization of any transient or event occurring in or calculated for a typical LWR is through a number of time trends (i.e. pressures, levels, temperatures, mass flow rates versus time). The main way to characterize the transient is through the event time, or the time elapsed since the beginning of the event, together with the initial and boundary conditions. In this case, which can be identified as ‘time domain’, time is taken as the horizontal axis in the graphical representation of the transient evolution. Therefore, in the area of uncertainty evaluation each transient becomes unique, thus requiring a specific evaluation of the error that characterizes any of the time trends. This is true notwithstanding the possibility of considering key phenomena or RTAs (see also Section 5.2) that are common to classes of transients.

A different way of looking at the same transients involves the use of ‘phase space’. In the graphical representation, any relevant quantity can be used on the vertical or horizontal axis. The comparison of the data of five experiments reproducing LB LOCA, SB LOCA and loss of feedwater scenarios in PWR simulators gives an idea of the differences between the time domain and phase space approaches [IV–1].

The basic idea underlying the CIAU is that any of the regions into which the phase space is subdivided can be assigned one uncertainty value. The same idea, with respect to specific thermohydraulic phenomena, is discussed in Refs [IV–2, IV–3], which show that phenomenological areas or regions in the phase space are suitable for use in scaling and extrapolation studies. Additional support for planning the method stemmed from the characterization of a generic plant state for the actuation of accident management countermeasures, as discussed in Ref. [IV–4]. Finally, the approach pursued is similar to that proposed by Groeneveld et al. [IV–5], where pressure, quality and flow rate are entered into a look-up table, which produces a suitable value for the CHF. In the present case, appropriate 'driving quantities' are entered into matrices and vectors and produce uncertainty values.

The concept of plant state is introduced in order to implement the aforementioned idea into the uncertainty evaluation process. Reference is made to any transient situation assumed to occur in BWR or PWR equipped nuclear power plants. No distinction is made between DBAs, beyond DBAs, operational transients or transients involving multiple failures. The only boundaries are constituted by the values assumed by the transient driving quantities considered. However, the hypothesis is made that the transients do not evolve towards situations that involve core degradation and loss of geometric integrity. It can be assumed that code validation must be proved within the fixed boundaries or ranges of variation of the assigned parameters.

For any plant transient scenario (i.e. SB LOCA, LB LOCA, transient or operational transient), the state of a plant can be characterized by six driving quantities and by the transient time. In the case of a PWR, the six quantities are: (a) the upper plenum pressure; (b) the primary loop mass inventory (including pressurizer); (c) the steam generator pressure; (d) the cladding surface temperature at 2/3 of the core active height (starting from the bottom of the active fuel), where the maximum value occurring in one horizontal core cross-section is considered; (e) the core power; and (f) the steam generator downcomer collapsed liquid level; if levels are different in the various steam generators, the largest value is considered. These are listed as a–f in Table IV–1. The transient time requires the specification of a 'zero' value ($t = 0$ s) starting from normal operating conditions. The hypothesis here is that a stable steady state (or stationary) situation must occur, or be specified in the case of a code calculation, before $t = 0$. If a BWR is considered, five driving quantities apply (i.e. all of the above except (c)). In this case, the quantity specified under (f) is the reactor pressure vessel downcomer level.

TABLE IV–1. SUBDIVISION OF DRIVING QUANTITIES INTO INTERVALS

(a) Upper plenum pressure (MPa)	(b) Primary circuit mass inventory (%)[a]	(c) Steam generator pressure (MPa)	(d) Cladding temperature at 2/3 core height (K)	(e) Core power (%)[a]	(f) Steam generator level (%)[a]
18.0	120	9.0	1473	130	150
15.0	100	7.0	973	100	100
10.0	80	3.0	643	50	50
9.0	40	0.1	573	6.0	0
7.0	10		473	1.0	
5.0			298	0.5	
4.0					
2.0					
0.5					

[a] Of the initial (nominal) value.

In relation to each of the driving quantities and the transient time, upper and lower boundaries must be fixed, and there should be a minimum optimal number of intervals. The assumed quantity and time related subdivision can be found in Tables IV–1 and IV–2, respectively. Six dimensions, namely (a–f) above, constitute the phase space domain (five in the case of a BWR). Each combination of intervals identifies one hypercube in that domain. Therefore, a hypercube and a time interval characterize a unique plant state within the framework of uncertainty evaluation. All plant states are characterized by a matrix of hypercubes and by a vector of time intervals.

The definition of time and quantity uncertainty can be obtained from Fig. IV–1. The dotted line is the result of a system code calculation: Y is a generic thermohydraulic code output plotted versus time. Each point value in the curve is affected by a quantity error (U_q) and by a time error (U_t). Owing to the uncertainty, each point value may take any value within the rectangle identified by the quantity and the time errors (Fig. IV–1(c)). The amount of error (i.e. each edge of the rectangle) can be defined in probabilistic terms, in consistency with the respective recommendations incorporated in current licensing approaches. The way to combine the rectangles at the end of the CIAU process is shown in Fig. IV–1(d).

TABLE IV–2. SUBDIVISION OF TRANSIENT TIME INTO INTERVALS
(applicable to a generic ITF or nuclear power plant transient)

No.	Transient duration (physical time) (s)	Period (s)	Time step (s)	Time interval[a]
1	0–100	0–100	1	1–100
2	0–1000	0–100	1	1–100
		100–1000	2	101–550
3	0–10 000	0–100	1	1–100
		100–1000	2	101–550
		1000–10 000	5	551–2350
4	>10 000	0–100	1	1–100
		100–1000	2	101–550
		1000–10 000	5	551–2350
		>10 000	10	2350–∞

[a] Used in TUV (see Table IV–3).

More specifically, the idea at the basis of the CIAU can be described as the uncertainty in code prediction being constant within each plant state. A QUM and a TUV can be set up, using the graphical representations given in Fig. IV–1. Additional considerations are provided below.

- The upper and lower limits of the driving quantities (Table IV–1) reflect either the physically allowed values or the boundaries of validation of system codes.
- The range of each interval in the quantity table (Table IV–1) and in the time vector (Table IV–2) is arbitrary. A decrease in the range means an increase in the number of intervals and an even greater increase in the number of hypercubes. The validity in the selection of these ranges can be verified a posteriori, when the QUM and the TUV are filled by data.
- Based on the intervals in Table IV–1, the total number of hypercubes equals about 8000. However, not all the combinations of intervals are realistic; for example, the combination of very low pressures and very high core power would be improbable. In practical terms this only means that some hypercubes will be outside the range of any transient and, most probably, will not add to the uncertainties.
- An LB LOCA of short duration (a few tens of seconds), an SB LOCA of long duration (several hundreds or even thousands of seconds) or transients of very long duration (up to ten thousand seconds), even

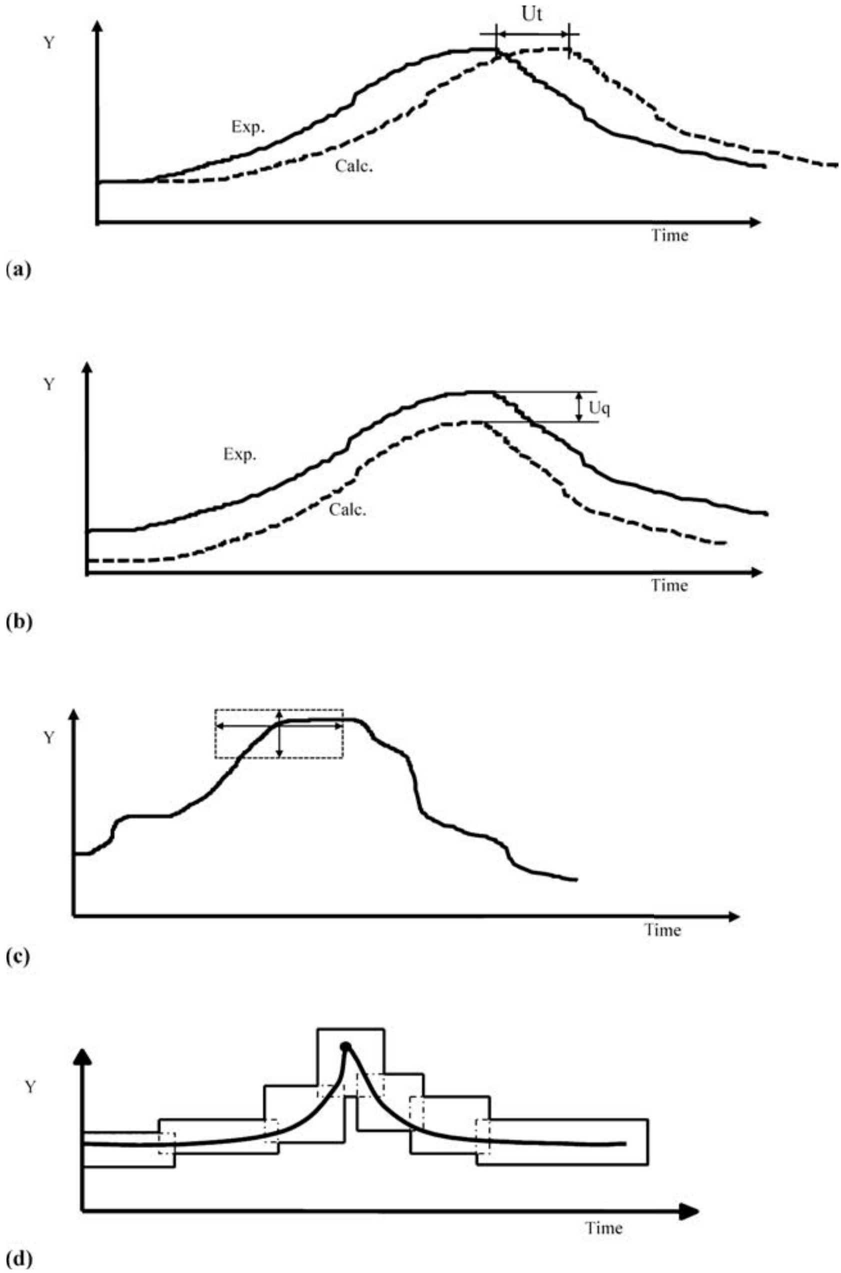

FIG. IV–1. Graphical representation of quantity and time errors to be included in the QUM and the TUV, respectively. (a) Only time error is present. (b) Only quantity error is present. (c) Combination of errors. (d) Derivation of continuous uncertainty bands.

without loss of primary loop integrity, produce quantity uncertainties that may affect the same hypercubes. However, the actual uncertainty that characterizes the values of a generic quantity during a transient of short or long duration is different because it is the combination of quantity and time values (Fig. IV–1). The error corresponding to the time value uncertainty is a ‘non-decreasing’ function. In the database assembled so far, no systematic differences between uncertainty values of different origins have been detected. Nevertheless, data from SB LOCAs, LB LOCAs, transients and operational transients that produce quantity uncertainty suitable for the CIAU QUM and TUV are clearly recognizable.

- Uncertainty data are continuously gathered and combined in line with the set-up and qualification of the CHF look-up table [IV–5]. When a reasonable number of data are available for each hypercube, the consistency in the selection of the hypercube range can be checked together with the hypothesis of mixing relevant data from SB LOCAs, LB LOCAs and transients.
- Each transient scenario in a nuclear plant evolves throughout a series of subsequent states. Each time the event passes through a hypercube and a time interval (i.e. a plant state), it assumes optimum uncertainty values. In this way, the entire event can be associated with uncertainty bands.

IV–3. CIAU PROCESS

The development of the CIAU requires a qualified system code (e.g. Ref. [IV–6]) and a suitable uncertainty methodology (e.g. Ref. [IV–7]). However, any of the available system codes or the uncertainty methodologies can be combined to obtain a code with internal assessment of uncertainty. A simplified flow diagram of the CIAU is given in Fig. IV–2, where two main parts are shown, the former dealing with the development of the method and the latter with its application.

The development of the CIAU benefited from the experience gained in the development of the UMAE uncertainty methodology [IV–7]. Many of the procedures used to propose the uncertainty method are also adopted here.

IV–3.1. Development of the CIAU

Development of the method requires the availability of qualified experimental data (block a in Fig. IV–2), qualified system code calculation results (block b), postulated transients, including the definition of plant states (block

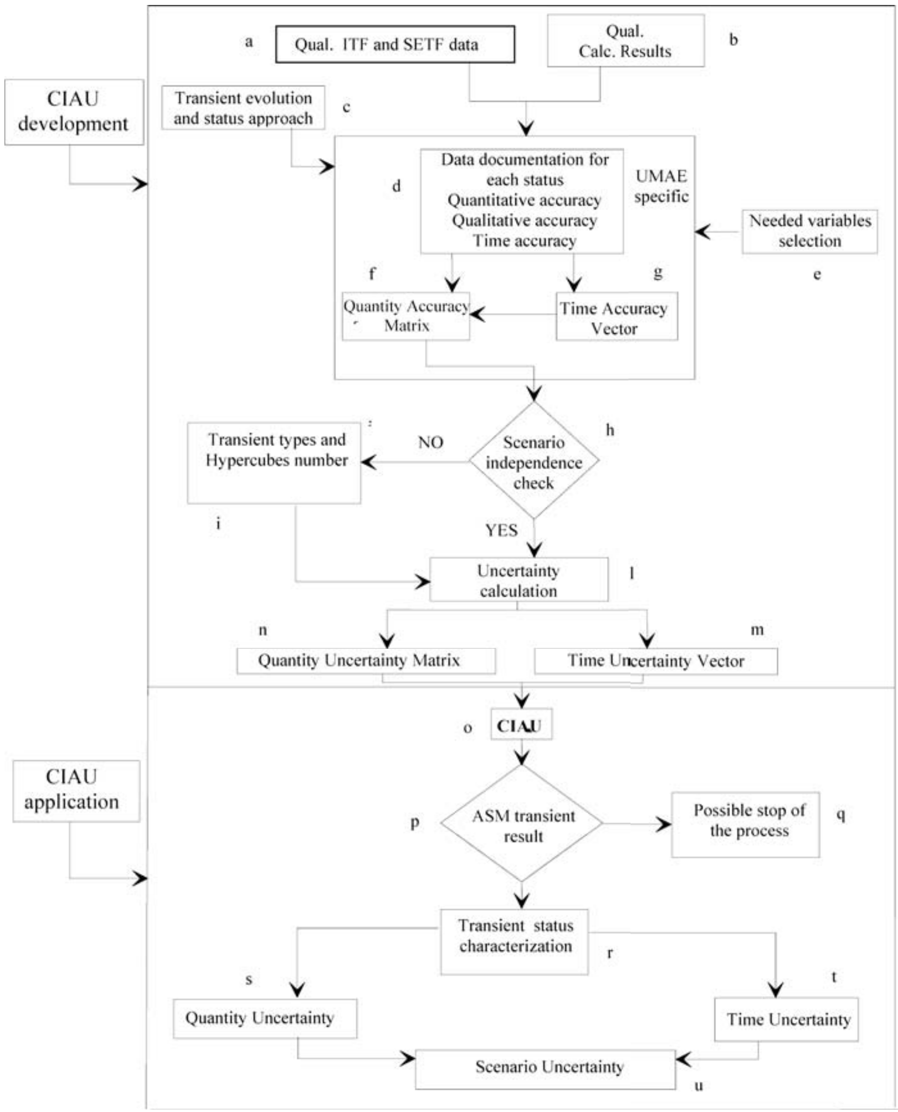

FIG. IV–2. Simplified flow diagram of the CIAU.

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, regardless of the type of qualification process. Qualified code results (block b) require the running of a qualified code on a qualified computer and compiler by a qualified user using a qualified nodalization [IV–8]. The qualification level of the code results is evaluated from a qualitative and a quantitative point of view, making use of the FFTBM in the latter case [IV–9].

Any uncertainty methodology supported by a system code can be used at block b for producing data that are related to block c, thus producing an uncertainty database. The output of a code calculation is thousands of variables, which are used to characterize a postulated transient scenario. It may prove to be impractical and unnecessary to evaluate the uncertainty connected with any quantity. Three variables have therefore been selected for uncertainty evaluation: the system pressure taken in the upper plenum of the main vessel, the (maximum) rod cladding temperature at 2/3 core active height and the fluid mass inventory in the primary circuit. It may be noted that the above quantities are the same as those used to characterize the plant state.

If the UMAE uncertainty methodology is used (bounded area in Fig. IV–2), relevant experimental data and code calculation results (blocks a and b) are compared. Accuracy is evaluated qualitatively and quantitatively (block d). If the accuracy is acceptable (block d), the quantity accuracy matrix (QAM) and the time accuracy vector (TAV) are generated (blocks f and g, respectively).

The various plant states identified under block c can now be filled by data from block b, or from blocks f and g in the case of the UMAE. The scenario independence check (block h) needs to verify that the transient type does not affect calculated uncertainties in each hypercube. For example, it might happen that data from the analysis of several SB LOCAs produce uncertainty values much greater than data from the analysis of a similar number of LB LOCAs, when the same hypercubes are concerned. In this case, the outlet 'NO' from block h leads into block i. The number of hypercubes (i.e. the ranges of variation of the driving quantities) must be changed or the transient type must be identified inside each hypercube. If the scenario independence check is concluded positively, uncertainty values can be meaningfully assigned to each plant state. The QUM and TUV are generated.

IV–3.2. Application and current status of the CIAU

Application of the CIAU is straightforward once the QUM and TUV are available. The 'error matrices' and the 'error vector' are currently used as a post-processor of a CIAU calculation. The ASM (i.e. a qualified nuclear power plant nodalization in the UMAE nomenclature) is used to obtain the transient scenario. Once a generic event is predicted (block p), the six driving quantities are used to identify the succession of hypercubes. The time intervals are also identified by the predicted event time (block r). This leads to the quantity uncertainty and time uncertainty values (blocks s and t, respectively), which

TABLE IV–3. COUPLES OF QUANTITY UNCERTAINTY MATRICES AND TIME UNCERTAINTY VECTORS DEVELOPED WITHIN THE FRAMEWORK OF THE CIAU

No.	Set of QUM and TUV	Objective	Reference database
1	CIAU goal	CIAU	UMAE qualified
2	CIAU extension	Code user effect and expansion of the database	UMAE qualified and available from the literature
3	CIAU test	Prove capabilities and flexibility of the method	Arbitrary data
4	CIAU R5/M2	Exploit the available database and constitute a reference	RELAP5/MOD2 SB LOCA related

can be combined to obtain the uncertainty bands sought. It may be noted again that uncertainty bands only envelop the quantities selected under block e. The computer tool UBEP is used to combine time and quantity uncertainty at each time step of the predicted event (block u). Continuous uncertainty bands are generated and envelop the ASM calculation results.

Within the framework of the development of the CIAU, four QUMs and four TUVs are considered. These are characterized in Table IV–3, where the objective for each set of coupled QUM and TUV is also given. The objective of the first set is to obtain the CIAU. Any calculation used in the process and the corresponding experimental database is qualified in the sense required by the UMAE. The second set has been included in order to enlarge the database that can be obtained through the UMAE. The following implications arise if the data are gathered from the literature:

(a) The nodalizations may not be qualified.
(b) User choices can be different from the standard choices required in the UMAE process; therefore the user effect accounts for an increasingly large part of the uncertainty value.
(c) The experimental data may not be qualified.
(d) No acceptability condition is fulfilled in the comparison between measured and predicted trends.
(e) The number of data points producing the QUM and TUV can be substantially greater than in the previous case (advantage of QUM and TUV set No. 2).

The third set of QUM and TUV has been created to test the numerical tools component of the CIAU, to prove the feasibility of the method and to demonstrate its capabilities. The uncertainty values have been arbitrarily assigned inside each hypercube and in relation to each time interval. The fourth set has been generated based on the wide experience gained and the resulting extensive database compiled from the application of RELAP5/MOD2 to SB LOCA analyses (e.g. Ref. [IV–10]). The objective is to apply uncertainty results obtained by the UMAE and related to the RELAP5/MOD2 code to calculations performed with the RELAP5/MOD3.2 code. The application field is restricted to SB LOCAs in PWRs. The availability of QUM and TUV set No. 4 permits a further qualification of set No. 1.

IV–3.3. The qualification processes

One important aspect of any tool developed in system thermohydraulics is the capability to perform an assessment, and possibly to show the quality level, using databases independent from those used in the development of the tool itself. Three qualification steps are foreseen in the case of the CIAU. All of these address QUM and TUV set No. 1 in Table IV–3.

The first step can be identified as the internal qualification process. Data gathered inside each hypercube or each time interval of the QUM and TUV, or inside the QAM and TAV if the UMAE methodology is adopted, are labelled before being combined. In other words, each uncertainty or accuracy related value includes its origin (i.e. the transient scenario type and the part of the hypercube concerned). A statistical analysis can be used to find out whether groups of data coming from different events or related to different parts of the same hypercube are different. If this is the case, different matrices of hypercubes must be built up separating the event types, and/or the dimensions of hypercubes in the phase space must be decreased. This process is continuously ongoing during the development of the method. The experience gained to date has not led to the need to increase the number of hypercubes nor to characterize the event type.

The second qualification step is carried out when a reasonable number of hypercubes and time intervals have been filled. In this case, the CIAU is run to simulate qualified transients measured in ITFs that have not been used to obtain uncertainty values. This step has been successful if the uncertainty bands calculated by the CIAU envelop the experimental data. It should be understood as the reference qualification process for the CIAU, together with the condition that the uncertainty bands be reasonably large. Through the completion of this step it will also be possible to establish the confidence level of the uncertainty statements on an objective basis. The increase in the number of positively completed qualification analyses will increase the confidence level of the procedure. No correlation has been established yet between the number of qualification analyses and the expected confidence level of the uncertainty results, although the target is to achieve a 95% confidence level.

The last qualification step is based upon the comparison of data gathered for QUM and TUV sets Nos 1 and 4 in Table IV–3. This is only related to phenomena of interest in connection with SB LOCAs in PWRs. This qualification step has been completed successfully if the predicted uncertainty bands are very similar in the two cases.

IV–4. TOOLS USED IN THE CIAU

In this section, the tools used in the CIAU are described and the self-standing tools (i.e. the RELAP5 system code and the UMAE uncertainty methodology), for which extensive literature exists, are distinguished from the numerical tools specifically developed within the framework of the CIAU project.

IV–4.1. Thermohydraulic system code

As mentioned above, the CIAU is based on the NRC version of RELAP5/MOD3.2 [IV–6]. The code solves six 1-D balance equations for mass momentum and energy, doing so separately for the steam and liquid phases. It has the capability to model any complex thermohydraulic system, including the primary loop, secondary loop and balance of plant systems in an LWR. Control systems can also be simulated together with the 0-D neutron kinetics performance of the core. The main reasons for the selection of the code may be stated as follows:

(a) Widespread use of the code (i.e. interest in using a RELAP5 based code on the part of the scientific community);
(b) Experience in its use at the University of Pisa, including the achievement of quality proofs (e.g. Ref. [IV–10]);
(c) Quality of the results produced, as demonstrated by various international organizations;
(d) Flexibility in developing nodalizations that also facilitates transferring to nuclear power plant expertise gathered from studying phenomena observed in ITFs.

IV–4.2. Uncertainty methodology

The UMAE uncertainty methodology [IV–7] is used in combination with RELAP5/MOD3.2 to produce the CIAU. The aim of this methodology is to calculate the uncertainty that characterizes the results of a thermohydraulic system code calculation. Among other things, it involves the fulfilment of different conditions of acceptability. Various steps in the method, including the use of statistics, are introduced in order to avoid expert judgement at any level in the process. Data from generic experiments in integral facilities and in SETFs, other than counterpart and similar tests, can be processed in the UMAE. One condition for the application of the method is that the plant scenario concerned, in relation to which uncertainty must be calculated, and the experimental database responsible for the accuracy of the code be alike.

A simplified flow diagram of the UMAE is given in Fig. IV–3. Some features of the methodology are discussed below.

IV–4.2.1. Nodalization development and qualification

Once the reference nuclear power plant event has been selected, related experiments in an ITF of a different scale must be identified. ITF nodalizations must be developed following standard guidelines. Experimental data must be compared with calculated data and both qualitative and quantitative accuracy must be evaluated. If the conditions and thresholds set for such an evaluation are fulfilled, the user of the methodology can exit by the path FG in Fig. IV–3 and may proceed with the development of the nuclear power plant nodalization. Again, this must be set up following the same guidelines as for the ITF. The qualification process has been completed positively if it can be shown that the nuclear power plant nodalization produces results that are in agreement with one of the ITF experiments (block k in Fig. IV–3).

IV–4.2.2. Definition of key phenomena and relevant thermohydraulic aspects

Key phenomena and RTAs are introduced within the framework of the evaluation of experimental and calculated databases in the UMAE process (see also Annex II). Key phenomena are attributed to a class of experiments. The lists prepared and agreed on by OECD/NEA–CSNI are used in the process (e.g. Refs [IV–11, IV–12]). RTAs are defined for a single transient and are characterized by numerical values of significant parameters. Around 20 RTAs, characterized by more than 40 values of significant parameters, must be selected for the qualitative evaluation of a database. Key phenomena and RTAs are used for the following purposes:

(a) To judge the relevance to scaling and the quality of a test facility (key phenomena);
(b) To judge the relevance to scaling and the quality of a test design (key phenomena);

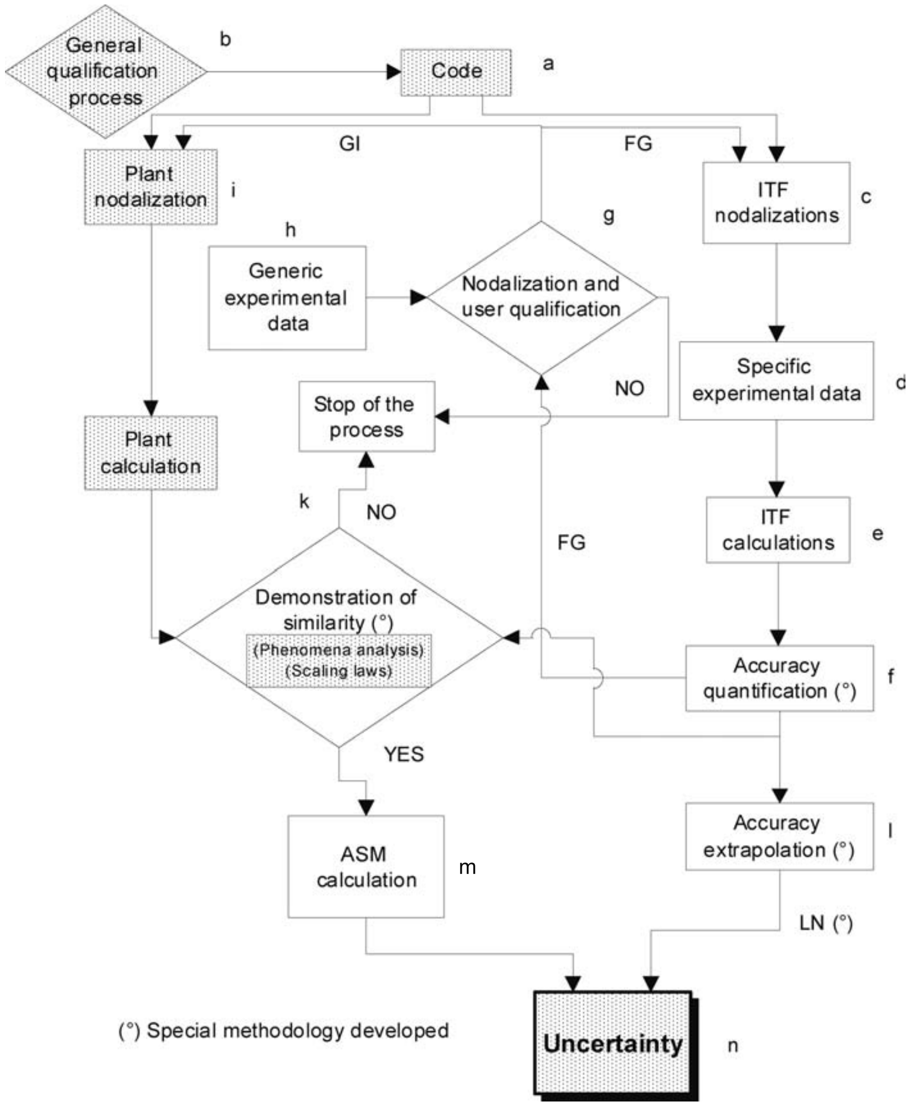

FIG. IV–3. Simplified flow diagram of the UMAE.

(c) To judge the relevance of an experimental database (key phenomena and RTAs);
(d) To judge the calculation performance (RTAs);
(e) To assess the success of a similarity study and of the nodalization qualification process (RTAs);
(f) To assess the similarity of different experimental databases (RTAs);
(g) To accept the ASM results before performing the accuracy extrapolation.

IV–4.2.3. Accuracy extrapolation

The accuracy in predicting the relevant ITF scenarios can be extrapolated if a number of conditions are satisfied. Important acceptability conditions are listed below (criteria exist for achieving any of the proposed goals).

(a) The design scaling factors of the ITF are suitable.
(b) The test design scaling factors are suitable.
(c) The experimental database is qualified.
(d) The nodalizations of the ITF and of the nuclear power plant are qualified at the steady state and at the 'on-transient' levels.
(e) The adopted code is generically qualified.
(f) Similarity of phenomena exists among the chosen experiments.
(g) Phenomena are well predicted by the code at a qualitative and a quantitative level. The meaning and implications of this statement can be understood from the description of the FFTBM.
(h) Phenomena are the same in the experiments and in the nuclear power plant related calculation. In other words, the results of the similarity study are acceptable.

The extrapolation of accuracy is achieved through the use of statistics [IV–13]. Numerical values, representative of the accuracy, are assumed to be randomly distributed around the ideal value. The large number of variables affecting the accuracy justifies this assumption. In this way, mean accuracy and 95th percentile accuracy are obtained, and are applicable to plant calculations. The measurement errors, the scaling distortions and the dimensions of the facilities involved are considered directly.

IV–4.2.4. Uncertainty calculation

A qualified plant nodalization becomes available through completion of the previous steps of the UMAE. This is known as the ASM. It may be noted that only one ASM calculation is necessary to complete the uncertainty evaluation process. This supplies the reference values that are used to extrapolate the accuracy. Different ASM calculations may be required if biases have to be introduced. Biases come from sources of uncertainties, if any, that are not present in the database used in the accuracy extrapolation process.

IV–4.3. Special numerical tools of the CIAU

The UMAE can be used as a 'black box' for generating uncertainty data suitable for filling the hypercubes and the time intervals of the CIAU. In this case, the UMAE ASM could be used to calculate several transients, and the resulting uncertainty bands could be transformed into the quantity and the time uncertainty values that constitute the QUM and TUV. This is not done within the framework under discussion for two main reasons, namely: (a) to keep track of the data generating the uncertainties; and (b) to make the qualification processes straightforward. This also saves person-months and computational resources. In particular, the results obtained by following the logical path FG in Fig. IV–3 are incorporated into the QUM and TUV. All the conditions for the acceptability of a calculation are preserved, as well as the possible interruptions (stops) in the process (blocks g and k) being taken into account. When performing the accuracy extrapolation, differences occur between the UMAE and CIAU processes, although the same formulas are used [IV–13]. In the former case, the extrapolation process is related to the accuracy data gathered from the analysis of a set of homogeneous transients measured in an ITF. In the latter case, the results from any kind of transient can be combined; the only condition is that they fall into the same hypercube (quantity accuracy) or time interval (time accuracy).

Finally, the UMAE ASM (block m in Fig. IV–3) is used to calculate reference nuclear power plant transients in the CIAU (block p in Fig. IV–2); the related qualification process is the same for both methodologies.

An overview of the tools and procedures adopted for the development of the CIAU, or which are necessary to implement the methodology, is provided in Table IV–4. The relevant procedures envisaged in the development or application processes have been incorporated in specific computer programs. The AFE, DAST and UBEP programs are described below.

IV–4.3.1. AFE tool

Before introducing the computer tool known as AFE, three areas of accuracy that cover the overall process, namely qualitative accuracy, quantification of accuracy and accuracy associated with the extrapolation process, are

TABLE IV–4. LIST OF PROCEDURES AND COMPUTER TOOLS REQUIRED TO DEVELOP AND RUN THE CIAU

No.	Procedure	Existence of software	Procedure adopted	Notes[a]
1	Selection of the nuclear power plant	No	—[b]	D and A
2	Selection of the reference nuclear power plant transient	No	—[c]	D and A
3	Nuclear power plant and ITF nodalization development	No	As in the UMAE	D and A
4	Nuclear power plant and ITF nodalization qualification at the steady state level	Yes[d]	As in the UMAE	D and A
5	Nuclear power plant and ITF nodalization qualification at the 'on-transient' level	Yes (FFTBM)[e]	As in the UMAE	D and A
6	Determination of accuracy data for the purpose of extrapolation	Yes (AFE)[f]	CIAU specific	D
7	Use of the statistical method	Yes (DAST)[f]	CIAU specific	D
8	Use of the ASM and performance of reference nuclear power plant calculation	No[g]	—	A
9	Determination of continuous uncertainty bands	Yes (UBEP)	CIAU specific	A
10	Introduction of biases if necessary	No	As in the UMAE	A
11	Interpretation of uncertainty results	No	—[h]	

[a] D: development of the CIAU; A; application of the CIAU.
[b] Must be consistent with the database.
[c] As above. This could be determined through a probabilistic safety assessment study.
[d] A table of threshold values is available.
[e] Including the demonstration of similarity foreseen by the UMAE process. This also involves possible interruption of the CIAU process.
[f] Only in the phase of development. This procedure is not used for running the CIAU.
[g] Any recommendations provided in the manual should be considered. Qualification as in the UMAE.
[h] This activity is connected with the follow-up and implications of the results obtained with the CIAU.

summarized in brief. In all cases, the accuracy is related to the comparison between measured and calculated time trends or quantities.

The qualitative accuracy starts with the visual observation of time trends. RTAs are introduced and characterized by digital values. Calculated and measured corresponding digital values are compared and evaluated qualitatively. The advantage of the process is to be able to show a one to one correspondence between RTAs in the experiment and in the calculation. In addition, the outcome of the calculation must show a reasonable agreement among the values of the relevant quantities. A positive result from the qualitative evaluation process is required before going on to the accuracy quantification.

The quantification of the accuracy is directed towards achieving the acceptability of any set of computer code calculated results. In this case, the FFTBM is used. This entails transformation into the frequency domain of the measured and predicted time trends of important variables. Acceptability thresholds are introduced that must be satisfied before any use of the above mentioned database can be made in the UMAE or in the CIAU processes.

If extrapolation is required, the accuracy of a generic calculation can be deduced through the use of the AFE tool. The quantity $A_j = |1 - Y_E/Y_c|$ is considered in the extrapolation process, where Y_c and Y_E are the values of a generic thermohydraulic quantity. Therefore quantity accuracy (QA) and accuracy in predicting time of phenomena within the transient (i.e. time accuracy (TA)) are obtained. QA and TA are evaluated in relation to any time interval, and considered separately in the measured and calculated data sets.

The list of events in Table IV–5 is used to characterize the time spans in the experimental and calculated databases. Any experimental database in an appropriate ITF or SETF combined with a code simulation can be used to produce QA and TA for the corresponding hypercube and time interval. Any thermohydraulic quantity calculated by the code and measured in the experiments is eligible to be considered for uncertainty evaluation in a hypercube. The upper plenum pressure, the rod surface temperature at 2/3 of core height and the mass inventory in the primary loop have been chosen here to fill the QUM. Transient time is necessary to fill the TUV. Although the variables selected for the QA coincide with three of the six driving quantities, this is not a prerequisite for the process. Assuming that experimental and calculated databases that fulfil the UMAE conditions[14] are available, the AFE tool completes the following steps:

(a) Determination of time spans on the basis of the events listed in Table IV–5. Time spans generally have a different duration in the experimental and calculated scenarios.

TABLE IV–5. LIST OF TIME EVENTS USED TO IDENTIFY COMPARABLE TIME SPANS IN THE EXPERIMENTAL AND CALCULATED DATABASES *(input into the AFE computer tool)*

Test start
Scram
Main steam line valve operation (closure, opening)
Main feedwater operation (closure, opening)
Pumps trip and coastdown limits
Blowdown in saturation condition
Pressurizer pilot operated relief valve actuation (start and end of cycling)
Steam generator steam relief valve operation (as above)
ECCS (accumulators, low pressure injection system, high pressure injection system) start and end of liquid delivery
Dryout occurrence (at 2/3 of the active fuel height)
PCT event (at 2/3 of the active fuel height)
Rewetting occurrence (at 2/3 of the active fuel height)
Actuation of relevant engineered safety features (pressurizer heaters, chemical and volume control system, residual heat removal, etc.)
Neutron power peaks during an ATWS
Test end

(b) Determination of the time sequence of hypercubes: each time span may belong to one or more hypercubes and to one or more time intervals.

(c) Calculation of QA and TA from the definition of A_j, inside each hypercube and time interval, respectively. In the case of QA, values at different points in time are considered; therefore, an average value and a standard deviation are obtained for A_j in each hypercube.

[14] Note: Requirements for performing the uncertainty evaluation are listed systematically in Refs [IV–7, IV–14]. The experimental database must come from a qualified ITF and from qualified boundary and initial conditions. The scaling problem must be addressed; the measured data should also be acceptable. The calculated database must be obtained with a qualified/frozen version of a code, adopting a qualified nodalization developed following specified rules. The comparison between experimental and calculated data must demonstrate that the qualitative and quantitative accuracy criteria have been fulfilled. This involves the use of the FFTBM tool.

IV–4.3.2. DAST tool

The results obtained from AFE are stored in hypercubes and time intervals. These are related to different facilities and different types of test, each of which is identified. Once a suitable number of data points are gathered in each hypercube or time interval, DAST performs the statistical evaluation, using the theoretical background and the process outlined in Ref. [IV–1]. No restriction has been placed on the number of data points: at least ten data points obtained from at least three differently scaled facilities must be available in each hypercube to make the statistical evaluation reliable.

Several accuracy values are transformed into one uncertainty value per hypercube and per time interval. The following formula is adopted:

$$U = (A + E_V + E_S + E_\sigma)\, x|R|$$

where

U is one side of the uncertainty band widths;
A is the extrapolated accuracy inside the hypercube;
E is additional errors coming from sources;
R is the reference value calculated by the code.

The term in parentheses constitutes the non-dimensional uncertainty and is directly available in the QUM and TUV. In the above equation, E_V, E_S and E_σ are additional contributions to the error generated, on account of the dimensions of the facility and the dispersion of accuracy inside each hypercube or time interval taken from a single experiment and from the combination of experiments, respectively. The term E_S results from the presence of several accuracy data in each hypercube due to the same experiment. This term is zero in each time interval.

In obtaining A, weighting factors have been used to account for:

(a) Scaling distortions of each facility (data from nuclear power plant measurements are given a weight equal to one);
(b) Measurement errors;
(c) Data dispersion caused by the accuracy averaging process in each hypercube or time interval (outputs from AFE).

The weights are specified by engineering judgement, which is part of the development process of the CIAU (and of the UMAE), but which must not be exercised during the application of the methodology. The impact of the selected values of the weighting factors upon the predicted uncertainty results has been evaluated: different sets of reasonable weighting factors do not lead to substantial changes in the uncertainty bands.

Demonstration of the fact that the quality of code predictions is not affected by the dimensions of the considered facility, or that the code is applicable for nuclear power plant studies, constitutes the scaling problem. This is not directly dealt with in the DAST computer tool. However, the research that led to the formulation of the UMAE and to the introduction of the nuclear power plant states supports the current approach [IV–15, IV–16]. The internal qualification process must be completed in order to provide a (reasonable) guarantee of the scaling capability of the assembled database.

The results of DAST constitute the QUM and TUV. In particular, QUM and TUV Nos 1 and 2 in Table IV–3 are generated by running this computer tool.

IV–4.3.3. UBEP tool

UBEP is the actual post-processor of the CIAU methodology. Uncertainty bands are superimposed on time trends representative of the selected nuclear power plant transient scenario. This is calculated by the ASM.

The six driving quantities output from the ASM are first used to identify the sequences of hypercubes and the time intervals that characterize the selected nuclear power plant transient scenario. Thus time and quantity uncertainties are known at every point in time during the transient. A rectangle can be built up for each transient time, as represented by block c in Fig. IV–1. This is related to one of the three quantities selected for uncertainty evaluation. The last operation performed by UBEP consists of finding the envelope of all the rectangles (block d in Fig. IV–1). In this way, continuous upper and lower uncertainty bands are generated in relation to upper plenum pressure, rod cladding temperature at 2/3 core height and fluid mass inventory of the primary loop.

IV–5. HYPERCUBES AND IDENTIFICATION OF TIME INTERVALS

Hypercubes are identified by a six digit number. The number of digits is the same as the number of driving quantities. Each digit varies between one and the number of intervals by which each driving quantity is subdivided (Table IV–1). The number ‘1’ identifies the ‘bottom’ parameter range in Table IV–1. The digits from left to right identify the driving quantities from (a) to (f) defined in Section IV–2. Therefore, the number 732121 characterizes the following ranges of parameters (from left to right):

(a) Upper plenum pressure between 10 and 15 MPa;
(b) Primary circuit mass inventory between 80% and 100% of the nominal value;
(c) Steam generator pressure between 3 and 7 MPa;
(d) Cladding surface temperature at 2/3 of core height between 298 and 473 K;
(e) Core power between 1% and 6% of the nominal value;
(f) Steam generator level (downcomer collapsed) between 0% and 50% of the nominal value.

Time intervals are identified as a function of the transient time, as shown in Table IV–2.

IV–5.1. Results obtained by AFE

AFE is applied to fill the sets of QAM and TAV that generate QUM and TUV Nos 1 and 2 in Table IV–3. As indicated in Section IV–4.3.2, the final values in the QUM and TUV are obtained through the application of DAST. With respect to set No. 1, the ITF experiments listed in Table IV–6 have been successfully analysed so far and the related accurate data have been obtained. For example, the hypercubes listed in Table IV–7 are crossed by the LSTF SB-CL-18 transient (test No. 10 in Table IV–6). The corresponding physical time and the time interval numbers are also listed in Table IV–7, in the first and last column of the table, respectively.

A further example of data processing can be found in Fig. IV–4. The calculated and experimental trends of the LOBI BL-44 test (test No. 3 in Table IV–6) are shown. The vertical lines represent the time spans that are characterized from the resulting sequence of events (list in Table IV–5). The time evolutions of the pressure accuracy and of the TA are reported in the same figure. The accuracy data constitute the A_j values.

A typical result obtained by AFE (for QUM and TUV No. 1) related to hypercube No. 732342 and to time interval No. 670 is given in Table IV–8. The databases, comprising the experimental transients and the related code calculation results, are identified. The standard deviation caused by the dispersion of the accuracy inside the hypercube can also be recognized from the data in Table IV–8.

TABLE IV–6. TRANSIENTS USED TO FILL QUM AND TUV SET No. 1 LISTED IN TABLE IV–3
(calculations performed by the RELAP5/MOD3.2 code)

No.	Facility or plant	Test	Type	Relevant secondary side conditions	Emergency system in primary side	Number of hypercubes involved	End of test (s)
1	SPES	SP-SB-04[a]	SB LOCA $A_r = 6\%$ of A_{max} in CL 100% power	No AFW Steam relief valves active	Accumulators and LPIS in CL	20	1637
2	SPES	SP-SB-03[a]	SB LOCA $A_r = 6\%$ of A_{max} in CL 10% power	No AFW Steam relief valves active	Accumulators and LPIS in CL	19	2034
3	LOBI/ MOD2	BL-44[a]	SB LOCA $A_r = 6\%$ of A_{max} in CL 100% power	No AFW Steam relief valves active	Accumulators and LPIS in CL	22	2350
4	LOBI/ MOD2	BL-34[a]	SB LOCA $A_r = 6\%$ of A_{max} in CL 10% power	No AFW Steam relief valves active	Accumulators and LPIS in CL	25	2400
5	LOFT	L2-5[a]	LB LOCA $A_r = 200\%$ of A_{max} in CL	Trip of FW and MSL	Accumulators HPIS and LPIS in CL	16	120
6	LSTF	SB-CL-21[a]	SB LOCA $A_r = 6\%$ of A_{max} in CL 10% power	No AFW Steam relief valves active	Accumulators in CL	24	2250

TABLE IV–6. TRANSIENTS USED TO FILL QUM AND TUV SET No. 1 LISTED IN TABLE IV–3 (cont.)
(calculations performed by the RELAP5/MOD3.2 code)

No.	Facility or plant	Test	Type	Relevant secondary side conditions	Emergency system in primary side	Number of hypercubes involved	End of test (s)
7	WWER	KZ1[b, c]	Operational transient Loss of a primary loop flow 72% power	FW flow controlled by core power	—	2	240
8	WWER	KZ2[b, c]	Operational transient Loss of a primary loop flow 52% power	FW flow controlled by core power	—	1	310
9	WWER	KZ3[b, c]	Operational transient Partial loss of feedwater 72% power	SG control valve stuck open	—	2	120
10	LSTF	SB-CL-18[b]	SB LOCA $A_r = 6\%$ of A_{max} in CL 10% power	No AFW Steam relief valves active	Accumulators in CL	16	900
11	LOBI/MOD2	BT-15/16[b, d]	LOFW 100% power BT-15 pumps running BT-16 pumps tripped	AFW activated	—	17	12 600

TABLE IV–6. TRANSIENTS USED TO FILL QUM AND TUV SET No. 1 LISTED IN TABLE IV–3 (cont.)
(calculations performed by the RELAP5/MOD3.2 code)

No.	Facility or plant	Test	Type	Relevant secondary side conditions	Emergency system in primary side	Number of hypercubes involved	End of test (s)
12	LOBI/ MOD1	A1-04[b]	LB LOCA A_r = 200% of A_{max} in CL	Early core power trip Accumulator in CL	—	20	80
13	LOBI/ MOD2	BT-17[b]	LOFW	Delayed activation of AFW	Fast upper plenum depressurization	20	6390
14	SPES	SP-SW-02[b]	LOFW	Delayed activation of AFW	No ECCS intervention	13	6600
15	LSTF	LSLW[b]	LOFW	Delayed activation of AFW	No ECCS intervention	14	11 004

[a] Full report exits.
[b] Summary report exists.
[c] FFTBM thresholds could not be checked.
[d] Data do not fully comply with FFTBM thresholds.

A_{max}: maximum cross-section area of a pipe connected to the pressure vessel; A_r: break (or rupture) area; CL: cold leg; FW: feedwater; LPIS: low pressure injection system; HPIS: high pressure injection system; MSL: main steam line; LOFW: loss of feedwater.

TABLE IV–7. HYPERCUBES AND TIME INTERVALS APPLICABLE TO THE LSTF SB-CL-18 TRANSIENT AS A FUNCTION OF PHYSICAL TIME

Time (s)	Hypercube (No.)	Time interval[a] (No.)
0–1	843453	1
2–3	733453	2–3
4–8	733353	4–8
9–18	733352	9–18
19–41	733353	19–41
42–46	723343	42–46
47–73	623343	47–73
74–120	523343	74–109
120–128	523333	110–114
129–138	523332	115–119
139–144	513332	116–122
145–156	513433	117–128
157–160	513333	129–130
161–196	513332	131–148
197–258	513233	149–179
259–414	413233	179–257
415–496	313233	258–298
497–800	213233	299–450

[a] The time interval is characterized in Table IV–2.

IV–5.2. Results obtained by DAST

DAST is used to combine the data points entering each hypercube and each time interval. One example of results from the application of DAST can be found in Table IV–8. This is related to the same hypercube and the same time interval mentioned above. The results from DAST show the importance of the *E* terms in obtaining the width of the uncertainty bands (see also Ref. [IV–1]).

IV–5.3. Current status of hypercubes and time intervals

Figures IV–5 to IV–8 provide an overview of the database related to the sets of QUM and TUV Nos 1, 3 and 4 (see Table IV–3). The abscissa of

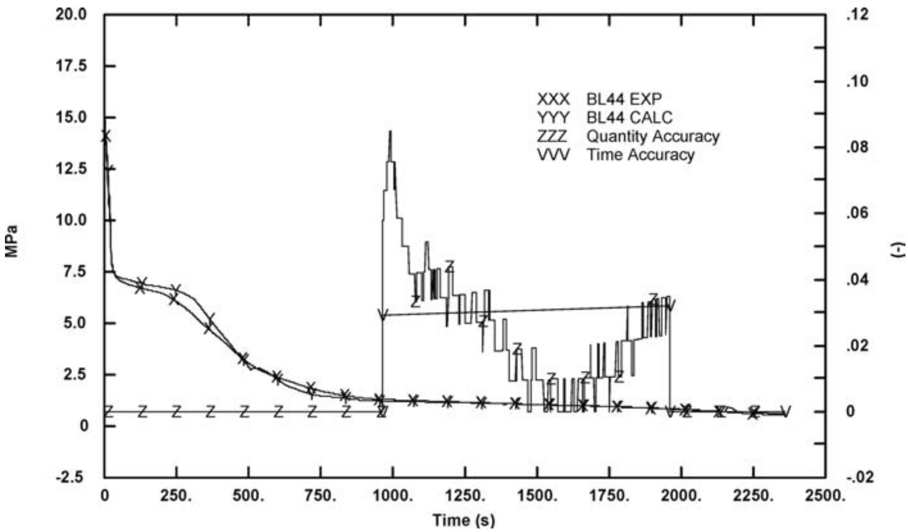

FIG. IV–4. Determination of pressure and time accuracy from the LOBI SB LOCA BL-44 test. The accuracy data are obtained from AFE and relate to one of the nine time spans (the time span from 964 to 1961 s) according to which the test has been subdivided.

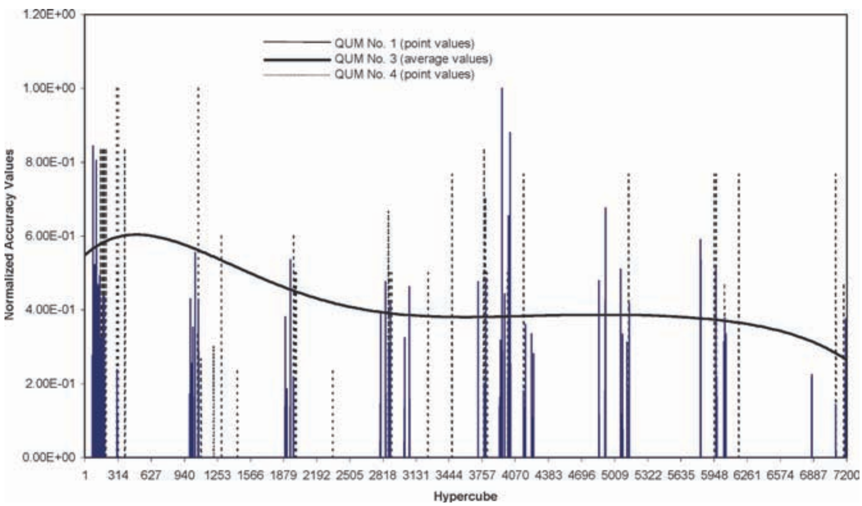

FIG. IV–5. Distribution of accuracy inside the hypercubes normalized to the maximum value for the set of QUM Nos 1, 3 and 4: primary system pressure.

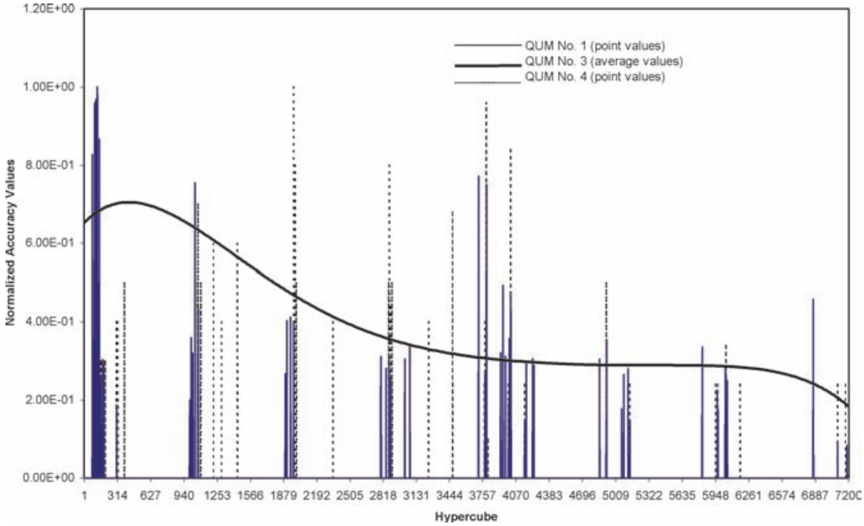

FIG. IV–6. Distribution of accuracy inside the hypercubes normalized to the maximum value for the set of QUM Nos 1, 3 and 4: primary system fluid mass inventory.

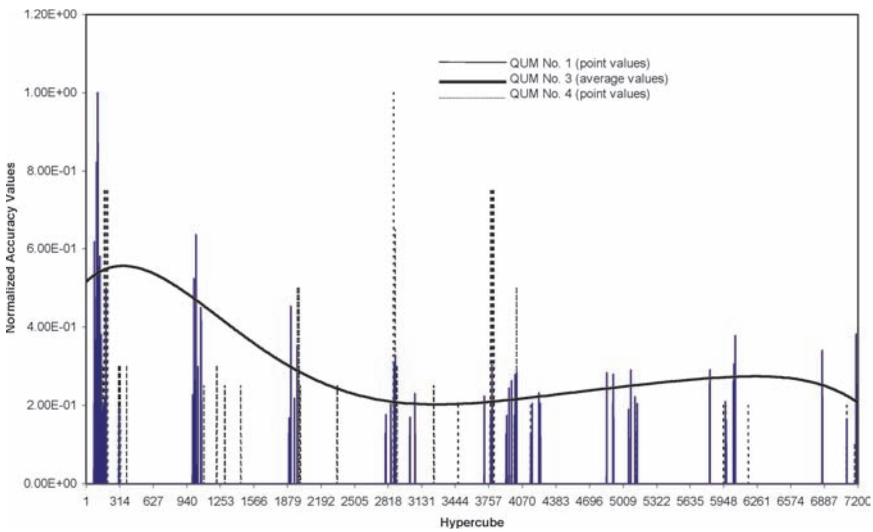

FIG. IV–7. Distribution of accuracy inside the hypercubes normalized to the maximum value for the set of QUM Nos 1, 3 and 4: fuel cladding temperature at 2/3 of core height.

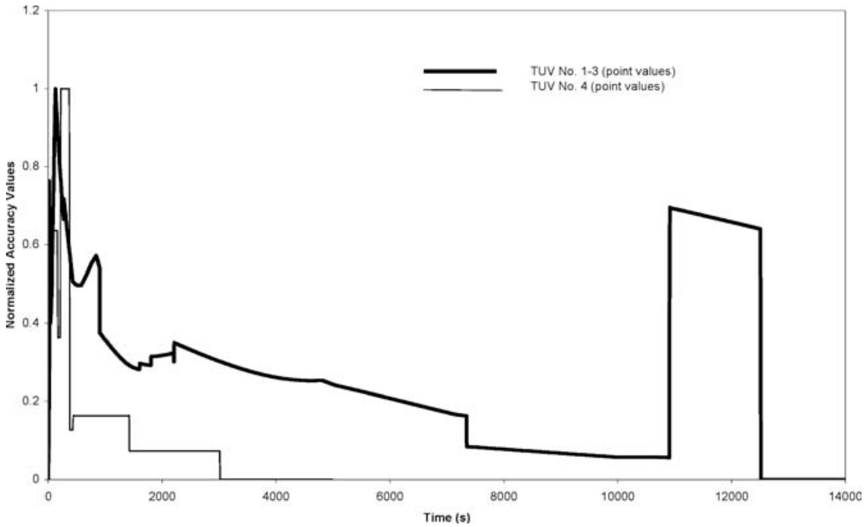

FIG. IV–8. Distribution of accuracy inside the time intervals normalized to the maximum value for the set of TUV Nos 1, 3 and 4. The correspondence between physical time and time interval can be found in Table IV–2.

Figs IV–5 to IV–7 is the sequential number of hypercubes that are reported from 1 to 7200 (i.e. hypercube No. 1 is the one identified as No. 111111 and hypercube No. 7200 is the one identified as No. 843553). The abscissa in Fig. IV–8 is the physical time during the transients. In this case, the data pertaining to TUV Nos 1 and 3 coincide: the existing TUV No. 1 data are assumed applicable for TUV No. 3. The following considerations apply:

(a) So far, a small number of hypercubes and time intervals include meaningful data, related to QUM and TUV set No. 1. This number is around 100, as can also be deduced from Table IV–6 (last but one column). The conditions for the application of DAST, namely at least three facilities and ten data points, are reached in a smaller number of hypercubes and time intervals.
(b) A large number of hypercubes, or plant states, when combined with time intervals, are not relevant or do not even occur in typical plant scenarios. This reduces the problem described under the item above.
(c) As mentioned in Section IV–3.2, QUM and TUV set No. 4 is obtained from RELAP5/MOD2 applications to SB LOCAs. The list of transients that generate the uncertainty can be found in Ref. [IV–1]. UMAE appli-

TABLE IV–8. RESULTS FROM THE APPLICATION OF AFE IN ONE GENERIC HYPERCUBE AND IN ONE GENERIC TIME INTERVAL *(development of QUM and TUV set No. 1 of Table IV–3; the transient that generates each data point can be identified)*

(a) AFE results[a]

Hypercube	Test ID No. in Table IV–6	Test ID in the computer tool	UP pressure accuracy	Mass inventory accuracy	Cladding temperature accuracy
732342	1	257SB04	0.043 0.0003 1.0	0.065 0.001 1.0	0.041 0.002 1.0
	3	712BL44	0.080 0.020 1.0	0.076 0.031 1.0	0.007 0.003 1.0
	4	712BL34	0.052 0.020 1.0	0.025 0.01 1.0	0.003 0.0006 1.0
	2	257SB03	0.060 0.006 1.0	0.030 0.004 1.0	0.007 0.001 1.0

Time interval	Test ID No. in Table IV–6	Test ID in the computer tool	Time accuracy[c]
670	1	257SB04	0.087 1.0
	3	712BL44	0.039 1.0
	4	712BL34	0.048 1.0
	2	257SB03	0.092 1.0
	13	712BT17	0.102 1.0
	14	257SP02	0.125 1.0
	6	48CL21	0.1385 1.0

TABLE IV–8. RESULTS FROM THE APPLICATION OF AFE IN ONE GENERIC HYPERCUBE AND IN ONE GENERIC TIME INTERVAL *(development of QUM and TUV set No. 1 of Table IV–3; the transient that generates each data point can be identified)*

(a) AFE results[a]

	10	48LSTF	0.138 1.0
	11	7121516	0.185 1.0

(b) DAST results[b]

Hypercube	UP pressure uncertainty	Mass inventory uncertainty	Cladding temperature uncertainty
732342	0.38	0.39	0.35

Time interval	Time uncertainty
670	0.49

[a] For each test, A_i, A_iS_i and P_{Di} values are reported.

[b] The DAST results are affected by the low number of data points and by the small dimensions of the considered ITF. The large value of the resulting U should be attributed to the E_V term.

[c] For each test, A_i and P_{Di} values are reported.

ID: identification; UP: upper plenum.

cations have been completed based on these transients. Uncertainty can therefore be evaluated for the SB LOCA class of transient.

(d) The set of QUM and TUV No. 3 is now ready to be used by the CIAU. In this connection, it should be pointed out once more that the reported accuracy values are not the result of AFE and DAST, but they have been arbitrarily fixed.

IV–6. SIGNIFICANT RESULTS ACHIEVED

The objective of the CIAU, as stated above, is to obtain uncertainty bands that bound results that vary with time, representative of transient scenarios in LWRs. These are the outcome of the application of a qualified system thermo-hydraulic code. Nevertheless, results related both to the application of the CIAU (i.e. associated with the above mentioned objective) and to the development of the same methodology are presented below. Results from the application of the adopted code (RELAP5) and uncertainty methodology (UMAE) can be found in Ref. [IV–14].

Significant results obtained when developing the CIAU are discussed here with the aim of providing a clear picture of the structure and the features of the methodology.

IV–6.1. Use of the CIAU

The code with the CIAU, available to the code user in its final configuration, has the same features as the original code, and no additional input requirement is needed to exploit the internal assessment of uncertainty capability. This capability is implemented as an automatic post-processor. In other words, each time the calculation of a nuclear power plant transient scenario is completed, the code automatically identifies the hypercubes through which the transient evolved. Each time the transient enters one hypercube (it is impossible to have a situation in which a scenario runs outside the hypercubes) quantity errors are picked up that combine with the time error at that particular point in time. Error bands are automatically generated and superimpose themselves on the calculated time trends of mass inventory, primary system pressure and rod surface temperature at 2/3 of the core active height. Therefore, no additional steps are required by the code user to run the original code if the CIAU is applied.

Attention, however, should be paid to ensuring consistency between relevant hypotheses (or conditions for code use) adopted during the development of the CIAU (to obtain the QUM and TUV) and those considered when running it. Consistency is achieved by taking the following into account:

(a) The range of parameters of the calculation must be consistent with the range of parameters used to create the QUM and TUV. For example, if (as is the case at present) the QUM and TUV are obtained for reactors equipped with a core constituted by cylindrical rods, the CIAU cannot be adopted for predicting uncertainty in a core consisting of flat fuel elements. The CIAU cannot be adopted for predicting transients in reactors operating at supercritical pressure.

(b) The nodalization must be qualified following the criteria and recommendations given in Ref. [IV–8]. These are adopted to obtain the QUM and TUV. In other words, one would not expect a wrong or unsuitable nodalization to produce correct error bands.

(c) The code version should be the same as the one adopted to obtain the errors (i.e. the QUM and TUV). An updated code version may be used provided it is demonstrated that the new version produces even better results than the (original) version adopted to obtain the errors. This may require analysis of selected transients in relation to which experimental data are available, together with results from the original code version.

IV–6.2. Results from the application of the methodology: A sample case

Reference results obtained from the use of QUM and TUV No. 3 are given here, mainly to provide an idea of the capabilities of the method. Owing to the manner in which QUM and TUV No. 3 have been obtained, the uncertainty bands presented below should not be considered as representative, from the quantitative point of view, of the results expected from the use of QUM and TUV No. 1.

The Krško two-loop Westinghouse reactor (about 650 MW(e)) constitutes the reference nuclear power plant. The list of transients that were calculated by RELAP5/MOD3.2 can be found in Ref. [IV–1]. The initial conditions correspond to the nominal conditions for the operation of the nuclear power plant. The nodalization adopted consists of about 300 hydraulic nodes and 2500 mesh points for conduction heat transfer.

The results of the use of UBEP, the final step of the CIAU process, are illustrated in Figs IV–9 to IV–13. In all cases, the thick line is the result of the ASM and the thin lines bound the predicted uncertainty.

IV–6.3. Results from the application of the methodology: Method qualification and safety studies

The CIAU methodology constitutes a pioneering effort to incorporate the CIAU into existing system codes. Although the methodology is mature, and has been used for pilot applications (see below), the embedded error databases (i.e. the QUM and TUV) need to be expanded to permit a possible full commercialization of the method.

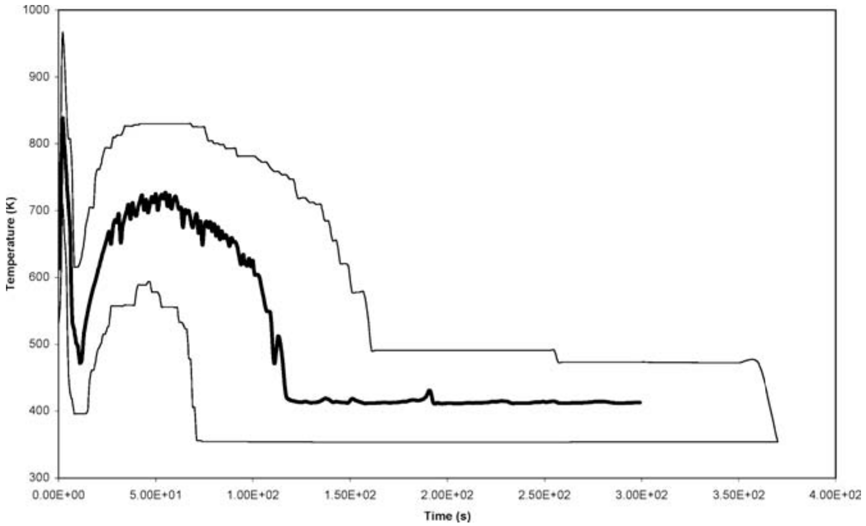

FIG. IV–9. Application of the CIAU to the analysis of an LB LOCA in a two-loop PWR: rod surface temperature at 2/3 of core height and related uncertainty bands.

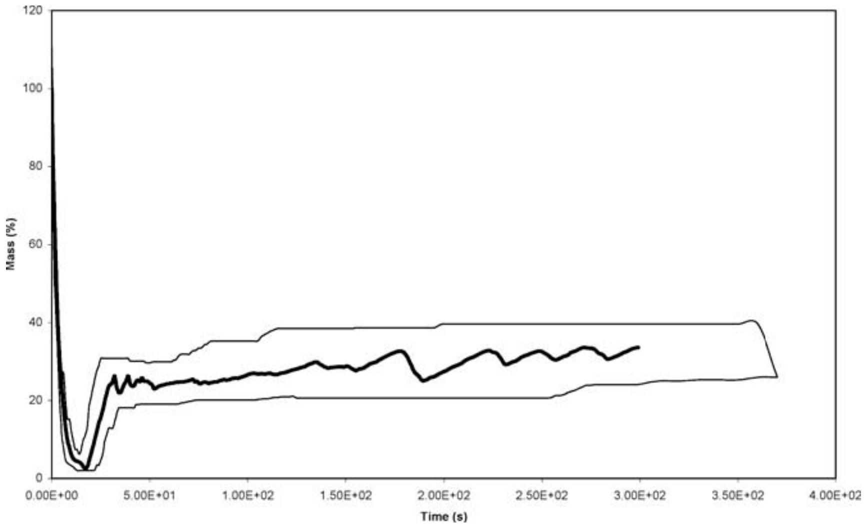

FIG. IV–10. Application of the CIAU to the analysis of an LB LOCA in a two-loop PWR: mass inventory in primary loop and related uncertainty bands.

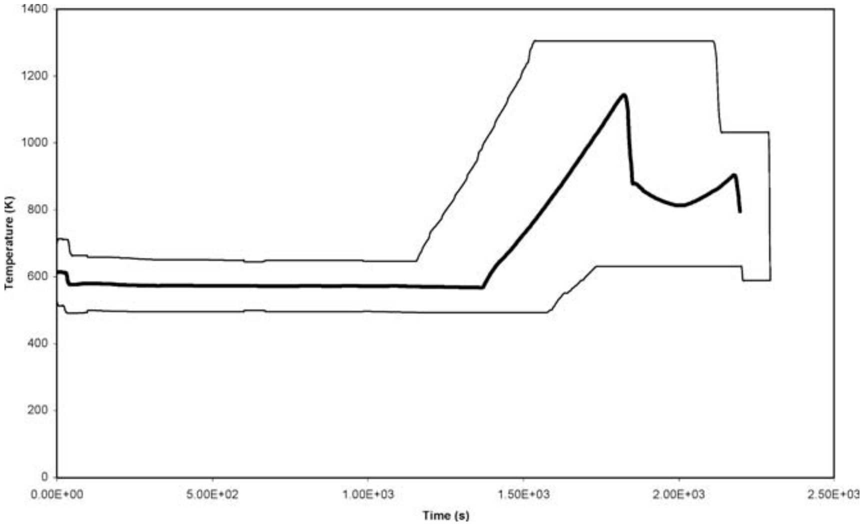

FIG. IV–11. Application of the CIAU to the analysis of an SB LOCA in a two-loop PWR: rod surface temperature at 2/3 of core height and related uncertainty bands.

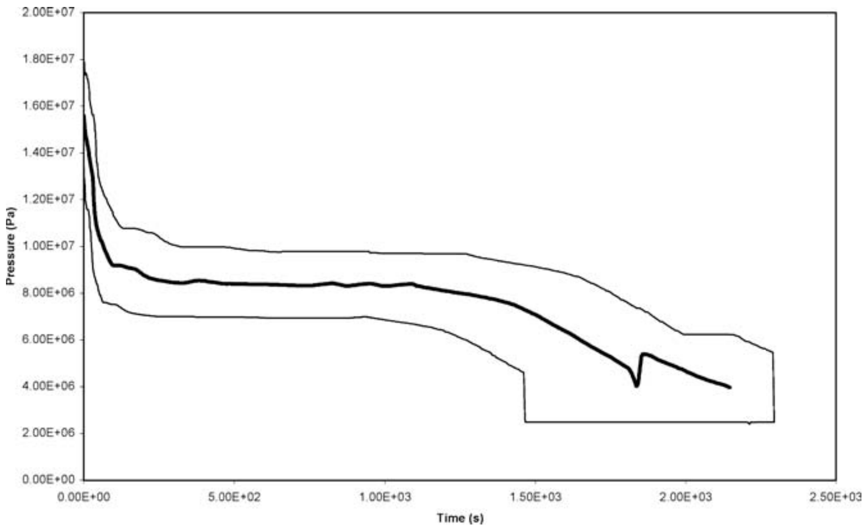

FIG. IV–12. Application of the CIAU to the analysis of an SB LOCA in a two-loop PWR: upper plenum pressure and related uncertainty bands.

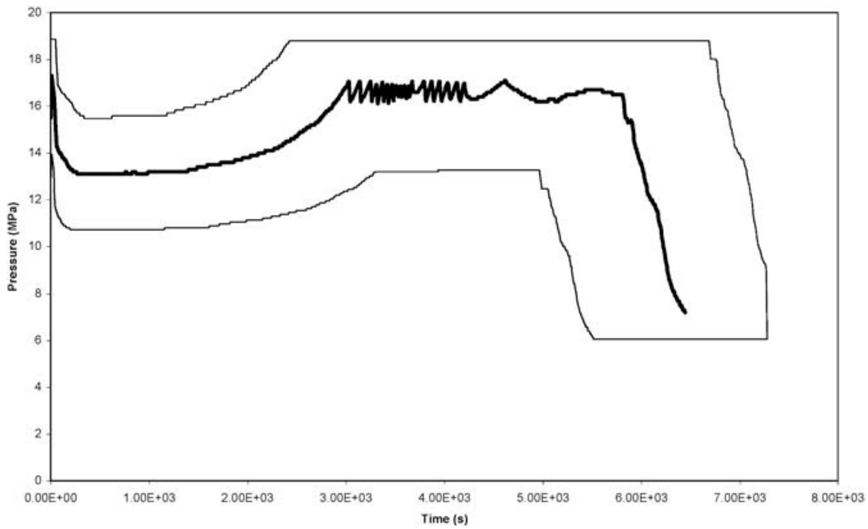

FIG. IV–13. Application of the CIAU to the analysis of an LOFW in a two-loop PWR: upper plenum pressure and related uncertainty bands.

IV–7. METHOD QUALIFICATION

‘Internal’ and ‘external’ qualification studies are part of the process of development of the CIAU method. Internal qualification studies are performed to confirm that data errors inside the QUM and TUV do not depend upon transient type, nuclear power plant type and hypercube dimensions. External qualification studies are carried out to demonstrate that a selected set of experimental (or nuclear power plant measured) data, not yet used as a source of errors for the QUM and TUV, lies between the upper and lower uncertainty bands associated with a BE prediction. Successfully completed external qualification studies deal, for example, with:

(a) An SB LOCA transient measured in the LSTF facility [IV–14];
(b) An SB LOCA transient measured in the LOBI facility [IV–17];
(c) An inadvertent load rejection transient at Angra 1 nuclear power plant [IV–18].

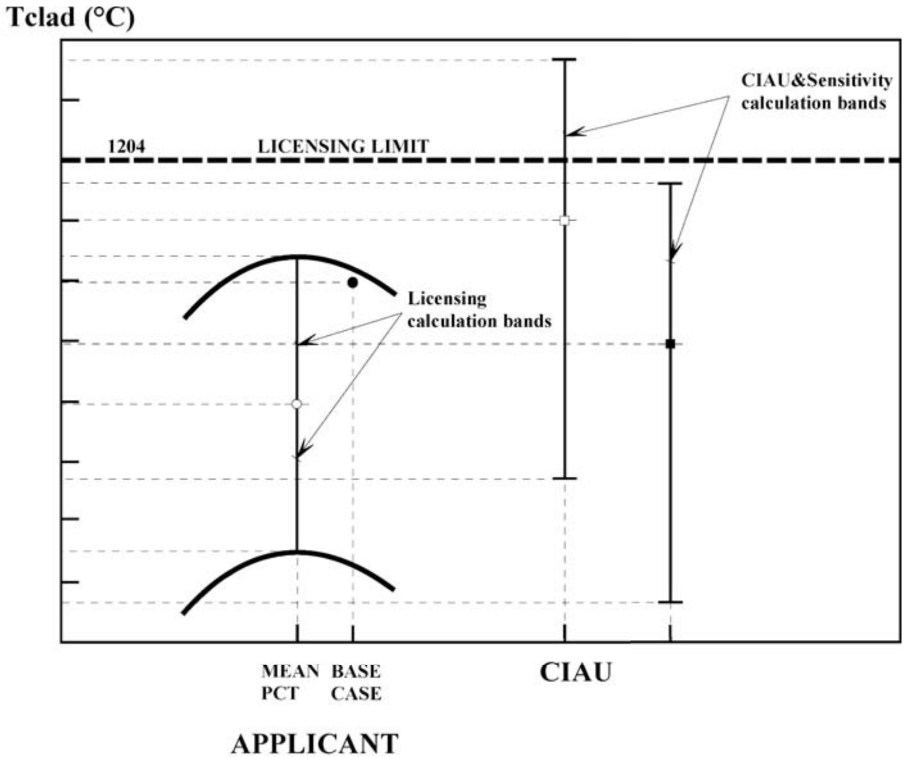

FIG. IV–14. BE plus uncertainty analysis of the LB LOCA DBA at Angra 2 PWR nuclear power plant: main result from the CIAU application [IV–19].

IV–8. SAFETY STUDIES

Two safety studies have been carried out with the CIAU method that are relevant for the nuclear industry. The results outlined below are taken from Refs [IV–19, IV–20], respectively, where further details can be found.

In the study described in Ref. [IV–19], the aim of the CIAU application was to perform an independent BE plus uncertainty analysis of the LB LOCA DBA at Angra 2 PWR nuclear power plant. The analysis is classified as ‘independent’ in the sense that it was carried out by different computational tools (code and uncertainty method) to those used by the utility applying for a reactor licence. The main results are summarized in Fig. IV–14, where the PCT and related uncertainty bands obtained by the CIAU and by the computational tools adopted by the applicant utility are given. The following comments apply:

(a) Continuous uncertainty bands were obtained by the CIAU method with regard to rod surface temperature, pressure and mass inventory in the primary system, but only point values for the PCT are considered in the figure.
(b) The CIAU (and the applicant's) analysis was carried out as a BE analysis. However, current rules for such an analysis might not be free of undue conservatism; the use of peak factors for linear power is the most visible example of this.
(c) The results of the CIAU method in terms of width of the uncertainty bands are very similar to those obtained by the applicant.
(d) The results of the CIAU method are supported by as many as 150 specifically oriented sensitivity studies (i.e. about 150 LB LOCA calculations have been performed to confirm the CIAU uncertainty results).
(e) The reference BE PCT calculated by the applicant (result on the left hand side of Fig. IV–14) plus the calculated uncertainty is lower than the allowed licensing limit of 1473 K.
(f) The reference BE PCT calculated by the CIAU (result in the centre of Fig. IV–14) is higher than the PCT 'proposed' by the applicant and is such as to cause the upper limit for the rod surface temperature to exceed the allowed licensing limit of 1473 K.
(g) It is shown that a lower BE PCT is calculated by the CIAU (result on the right hand side of Fig. IV–14); however, based upon the expertise available, including supporting evidence from experimental data, it has not been possible so far to justify user choices leading to such a result.

In the second study [IV–20], a code to code comparison problem was proposed related to the 200 mm LOCA analysis in a WWER-440. The objective of the analysis was to demonstrate that the results of predictions by two advanced computer codes such as RELAP5 and CATHARE are not in contradiction. In order to be able to demonstrate this, a (not necessarily BE) calculation of the above mentioned transient scenario with assigned hypotheses was performed by the RELAP5 code (Fig. IV–15). Uncertainty bands (the thick lines in Fig. IV–15) for such a calculation were obtained by the CIAU. The aim was to show that the CATHARE results would be embedded within the uncertainty bands of RELAP5 when the same transient was calculated with the same boundary and initial conditions. The CATHARE results shown in Fig. IV–15 are actually bounded by the uncertainty bands of the RELAP5 CIAU calculation, thus allowing a successful solution to the assigned problem.

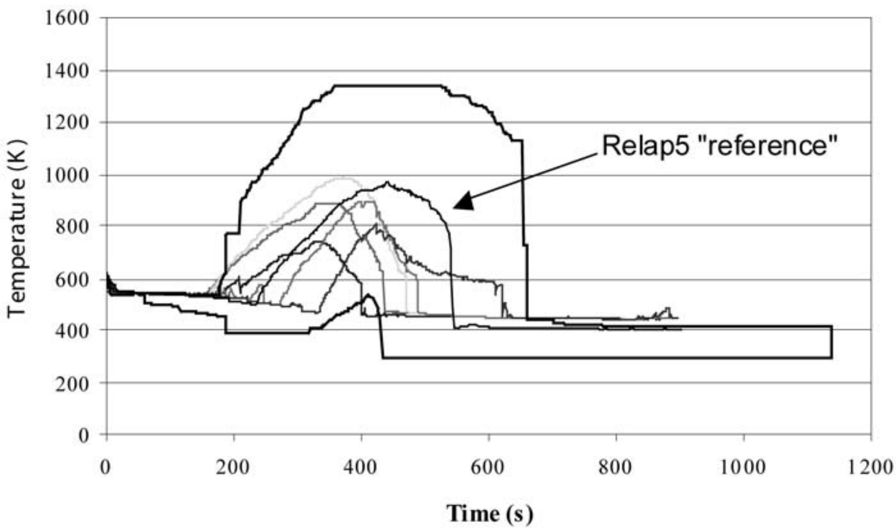

FIG. IV–15. Uncertainty analysis of the 200 mm LOCA DBA at the Kozloduy 3 WWER-440 nuclear power plant: main result from the CIAU application [IV–18].

IV–9. CONCLUSIONS

The CIAU constitutes a powerful tool that is obtained through the combination of a qualified BE thermohydraulic system code and a suitable uncertainty methodology. Reference is made to the prediction of a transient scenario as the consequence of a postulated event in a generic LWR. Implementation of the CIAU capability allows error (uncertainty) bands coupled with the time dependent results of the system code calculation concerned to be obtained. The experience gained in the application and the qualification of system codes (i.e. RELAP5, in this instance) as well as in the development and qualification of an uncertainty methodology (i.e. the UMAE) has been fully utilized within the present framework [IV–15, IV–16].

The idea at the basis of the CIAU is connected with the plant state approach. First, quantities have been selected to characterize, in a multidimensional space, the thermohydraulic state of an LWR during any transient. In this way, hypercubes have been defined and associated with time intervals accounting for the transient duration. The accuracy of each hypercube and time interval has then been calculated from analysis of experimental data. When applying the method, the combination of accuracy values obtained from hypercubes and time intervals permits continuous uncertainty or error bands to be obtained that envelop any time dependent variables that are the output of a system code calculation.

The RELAP5/MOD3.2 system code and an uncertainty methodology, the UMAE, have been coupled to constitute the CIAU. Thus the uncertainty has been obtained through extrapolation of the accuracy resulting from the comparison between code results and relevant experimental data. These data may be obtained from ITFs as well as from SETFs. However, any qualified system code and any uncertainty methodology could be applied in a similar manner to that described here to achieve the same goal.

A consistent ensemble of uncertainty values is included in any set constituted by a QUM and TUV. The QUM is formed by hypercubes, the edges of which consist of six selected variables representative of a transient scenario. The TUV is formed by time intervals. Four sets of QUM and TUV have been considered. To fill set No. 1 is the aim of the CIAU based research. The application of set No. 3 has been used in this report to show the features of the method. The results obtained prove the feasibility of the idea and the capabilities of the CIAU. The main advantage of the methodology lies in avoiding the need to interpret logical statements that are part of uncertainty methods (i.e. avoiding the user effect when using uncertainty methodologies). In addition, only negligible computer time or human resources are needed for the application of the CIAU.

The following aspects of the proposed methodology require additional investigation:

(a) Demonstration that accuracy results obtained from the analysis of different types of transients are statistically homogeneous inside each hypercube or time interval. This may require a finer subdivision of hypercubes or, in an extreme case, the setting up of different databases in accordance with the transient type.
(b) Qualification of the methodology. This can be achieved through the use of experimental data that are independent of those adopted when setting up the accuracy databases.
(c) Introduction of consistency checks between uncertainty values characterizing different variables (e.g. pressure and related saturation temperature).
(d) Introduction of the capability to consider bifurcation. In this case, any selected transient can be calculated to evolve through a tree of transients, each characterized by suitable uncertainty bands.

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SYMBOLS USED IN ANNEX IV

AA	average accuracy resulting from the FFTBM
AA_{tot}	average accuracy related to a group of variables (or time trends)
A_i	accuracy used in the AFE: this is the average of A_j
A_j	accuracy used in the AFE: this is the point value
E_S	contribution to uncertainty due to the spread of accuracy data in each hypercube
E_V	contribution to uncertainty due to the dimensions of the facilities generating the accuracy data
E_σ	contribution to the uncertainty generated by data combination in hypercubes and time intervals
K	acceptability threshold for accuracy used in the FFTBM
M	number of values of A_j
N	number of experimental (and corresponding code calculation) data sets
N_{var}	number of variables (time trends) used in the FFTBM
P_{Di}	DAST weighting factor of any A_i (dispersion of A_i inside the hypercube)
P_{ki}	DAST weighting factor of any A_i accounting for geometrical distortions of facilities generating the data

P_{si}	DAST weighting factor of any A_i accounting for the dispersion of A_j
R	reference results of the code calculation (the ASM is used)
S_i	dispersion associated to each A_i
U	one side of the uncertainty band
U_{t}	uncertainty band size (time dimension)
U_{q}	uncertainty band size (quantity dimension)
w_{f}	FFTBM weighting factor (overall)
WF	weighted frequency
$\mathrm{WF}_{\mathrm{tot}}$	weighted frequency related to a group of variables (time trends)
Y_{c} or Y_{calc}	generic calculated thermohydraulic quantity
Y_{E} or Y_{exp}	generic measured thermohydraulic quantity
σ	standard deviation in the distribution of P_iA_i Laplace transform

Annex V

EXAMPLES OF LICENSING APPLICATIONS

As both the results of a survey carried out by the OECD/NEA–CSNI [V–1] and the findings of an IAEA Technical Meeting that took place in 2005[15] have shown, regulations in most countries permit the use of BE codes. Examples of the application in the licensing processes of BE methods including uncertainty evaluation are provided below. (If not stated otherwise, all completed analyses were performed for postulated LB LOCAs.)

(a) Brazil: The Angra 2 nuclear power plant licensing analysis. The application was performed by Siemens [V–4] (now Framatome ANP) and reviewed by GRS. This constituted the first such application in the licensing process of a new nuclear power plant.
(b) Canada: Atomic Energy of Canada Ltd and other partners in the Canadian nuclear industry have developed the BEAU methodology for potential use in licensing applications for CANDU reactors. The BEAU methodology is consistent with the CSAU methodology developed in the early 1990s in the USA. Guidelines and applications for large and medium break LOCA calculations were reviewed by a panel of international experts [V–5, V–6, V–7].
(c) France: The utility EDF presented a statistical method (95% probability statement) using a response surface to substitute the BE code, plus a deterministic realistic calculation enveloping the 95% values [V–8]. The Institut de radioprotection et de sûreté nucléaire (IRSN), formerly the IPSN, reviewed this method for the French safety authority. The categorization of some key parameters into 'macroparameters' was not accepted, but the envelope based on representative integral tests was. A BE code with specific conservative models was used to bound these test results. The IRSN uses Wilks' formula to evaluate the uncertainty, relying on actual code results without approximations by fitted response surfaces, similarly to the GRS method.
(d) Germany: Efforts are under way to include realistic analyses and probabilistic uncertainty evaluation in the licensing regulations. In July 2005, the German Reactor Safety Commission issued a recommendation to perform uncertainty analysis in LOCA safety analyses. A more general

[15] Technical Meeting on Use of a Best Estimate Approach in Licensing with Evaluation of Uncertainties, 12–16 September 2005, Pisa.

requirement is included in a draft revision of the German nuclear regulations issued by the German Federal Ministry for Environment, Nature Conservation and Nuclear Safety.

(e) Republic of Korea: A BE evaluation of ECCS performance was developed by the Korea Electric Power Institute (KEPRI); the KEPRI realistic evaluation model (KREM) has been approved and several applications using KREM have been attempted. Examples are the application of KREM in connection with power uprates of present nuclear power plants and the safety issue of direct vessel injection ECCS of the Republic of Korea's APR-1400 advanced power reactor. The realistic evaluation methodology for power increase essentially follows the CSAU method, but using Wilks' formula (proposed by GRS) and performing 59 computer code runs to obtain 95% probability statements.

(f) Lithuania: The Lithuanian Energy Institute performed uncertainty analyses for the licensing process of the Ignalina unit 2 nuclear power plant (RBMK-1500), using the GRS method, investigating, for example, the group distribution header blocking event.

(g) Netherlands: Applied in the licensing process for a nuclear power plant upgrade and renewal of the licence for the Dodewaard nuclear power plant. The application was performed by General Electric and reviewed by GRS.

(h) USA: Applied in connection with updates to the final safety analysis reports of approximately 20 plants. These applications were performed by Westinghouse using the CSAU method [V–2] and response surfaces. The NRC's main aim was to investigate the compensation of code errors, the propagation of uncertainties and scalability. The CSAU method was also applied by Westinghouse to a 600 LB LOCA analysis [V–3]; the application was approved by the NRC.

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ABBREVIATIONS

AA	average amplitude
AEAT	AEA Technology
AEAW	Atomic Energy Authority Winfrith
AHP	analytical hierarchical process
ASM	analytical simulation model
ATWS	anticipated transient without scram
BE	best estimate
BEAU	best estimate and uncertainty
BEPU	best estimate plus uncertainty
BWR	boiling water reactor
CANDU	Canadian deuterium uranium (reactor)
CFD	computational fluid dynamics
CFR	Code of Federal Regulations (USA)
CHF	critical heat flux
CIAU	code with the capability of internal assessment of uncertainty
CSAU	code scaling, applicability and uncertainty
CSNI	Committee on the Safety of Nuclear Installations
DBA	design basis accident
DRM	deterministic realistic method
ECCS	emergency core cooling system
EDF	Electricité de France
ENUSA	Empresa Nacional del Uranio, SA
FFT	fast Fourier transform
FFTBM	fast Fourier transform based method
GRS	Gesellschaft für Anlagen- und Reaktorsicherheit
GSUAM	generic statistical uncertainty analysis method
IET	integral effects test
IPSN	Institut de protection et de sûreté nucléaire
ITF	integral test facility
LB	large break
LHS	Latin hypercube sampling

LOCA	loss of coolant accident
LSTF	Large Scale Test Facility (Japan)
LWR	light water reactor
NRC	Nuclear Regulatory Commission
OECD	Organisation for Economic Co-operation and Development
OECD/NEA	OECD Nuclear Energy Agency
OSE	optimal statistical estimator
PCT	peak cladding temperature
PDF	probability density function
PIRT	phenomena identification and ranking table
PTS	pressurized thermal shock
PWR	pressurized water reactor
QA	quantity accuracy
QAM	quantity accuracy matrix (or matrices)
QUM	quantity uncertainty matrix (or matrices)
RTA	relevant thermohydraulic aspect
SB	small break
SET	separate effects test
SETF	separate effects test facility
TA	time accuracy
TAV	time accuracy vector
TUV	time uncertainty vector
UMAE	uncertainty methodology based on accuracy extrapolation
UMS	uncertainty methods study
UPTF	Upper Plenum Test Facility
UVUT	unequal velocities, unequal temperatures
WWER	water moderated, water cooled power reactor (Russian design)
WF	weighted frequency

DEFINITIONS

These definitions were compiled solely for the purposes of this report. They do not represent a consensus or an endorsement by the IAEA. For further definitions the reader is referred to the IAEA Safety Glossary: 2007 Edition.

accident analysis. In its broad sense, as used in this report, the term is used for deterministic safety analysis of anticipated operational occurrences, design basis accidents and beyond design basis accidents.

accuracy. The known bias between a code prediction and the actual transient performance of a real facility.

best estimate analysis. Accident analysis which:

(a) Is free of deliberate pessimism regarding selected acceptance criteria;
(b) Uses a best estimate code;
(c) Includes uncertainty analysis.

best estimate code. A code which:

(a) Is free of deliberate pessimism regarding selected acceptance criteria;
(b) Contains a sufficiently detailed model to describe the relevant processes required to be modelled.

bias. Measure of the systematic difference between an actual or true value and a predicted or measured mean. Bias is the tendency of a model to overpredict or underpredict.

code. Numerical tool that is capable of predicting a physical phenomenon. In this report, the term ‘code’ is mainly used to mean ‘system code’ (see the related definition).

confidence level. In the general context of confidence intervals:

- Probability β that the confidence interval to be computed from the sample will contain the true parameter value.

In the context of statistical tolerance limits:

- Probability β that the limits to be computed will cover the specified proportion α of the population (probability content α). The confidence level is specified to account for a possible sampling error due to the limited sample size, for example a limited number of calculations, from which the statements are obtained.

conservative analysis. Analysis leading to pessimistic results relative to a specified acceptance criterion.

controlled safe state. A plant state in which:

(a) The core is and remains subcritical;
(b) The core is in a coolable geometry and there is no further fuel failure;
(c) Heat is being removed by the appropriate heat removal systems;
(d) Fission product releases from the containment have ceased, or further release can be bounded.

core damage. Substantial loss of the core geometry with major radioactive release, leading to conditions beyond the criteria established for design basis accidents, typically due to excessive core overheating.

evaluation model. A nuclear plant system computer code or any other analysis tool designed to predict the aggregate behaviour of a reactor during a loss of coolant accident. It can be either best estimate or conservative and may contain many correlations or models.

hypercube. Multidimensional solid defined in the phase space, where the phases are suitable leading thermohydraulic quantities (pressure, temperature, mass inventory, power, etc.) that allow the identification and characterization of a transient scenario in a nuclear power plant.

input deck. An input deck contains all input data necessary to run a computer code (job). These data have to be provided by the user of the code. The user has to select the models and options that are available, how the facility to be analysed is to be represented (nodalization) and the initial and boundary conditions of the accident or transient to be calculated.

integral test facility (ITF). Experimental loop designed according to a proper set of scaling laws, intended for the simulation of an entire nuclear power plant. It includes all the main components and geometrical zones of a nuclear power plant. Phenomena measured in an ITF are expected to be as similar as possible to those expected in the reference plant. (The design of an ITF is always based on a reference plant that has already been built or designed.)

internal assessment of uncertainty. Capability of a (thermohydraulic system) code to associate continuous uncertainty bands to the time dependent output of a code calculation.

nodalization. The geometric representation of a nuclear reactor system to be calculated by a computer code.

All major safety codes follow the concept of 'free nodalization' to allow flexibility. The user has to create a detailed noding diagram that represents the system. The code offers a number of basic elements, such as single volumes, pipes, branches, junctions and heat structures. Considerable responsibility rests with the user: nodalization is always a compromise between the desired degree of resolution and an acceptable computational effort. Continuous reduction of cell sizes does not automatically improve the accuracy of the calculation. Most empirical constitutive relations in the codes have been developed on the basis of a fixed (in general coarse) nodalization. Numerical schemes generally include artificial viscosity to provide stable numerical results; a reduction of cell sizes might result in non-physical instabilities.

Multidimensional effects exist even in small scale test facilities to be represented by a one dimensional code (i.e. at flow splitting and flow merging, additional bypass flows and large redistribution of flow during the transient).

phenomena identification. The process of subdividing a complex system thermohydraulic scenario (depending upon a large number of fundamental thermohydraulic quantities) into simpler components or phenomena that depend mainly upon single (or a limited number of) thermohydraulic quantities.

phenomenological window. Time span within a complex thermohydraulic scenario where a single phenomenon (or a limited number of phenomena) occurs.

probability density function, cumulative distribution function. The cumulative distribution function $F(x)$ of a variable X gives, for all values x, the probability that the variable X will be less or equal to x (i.e. $F(x) = P(X \leq x)$).

The probability density function $f(x)$ is the derivative of the cumulative distribution function of a variable (i.e. $f(x) = F'(x)$).

probabilistic uncertainty statement. Probabilistic statement about the uncertainty of the model outcome resulting from the uncertainties of the input parameters, usually formulated in the form of statistical tolerance limits.

reference calculation. Calculation using nominal, best estimate input values and default values for the computer code options and input data for models.

response surface. The result of techniques used in the empirical study of relationships between one or more responses and a group of input variables.

safety margin (absolute terms). The difference, in physical units, between the critical value of an assigned parameter associated with the failure of a system or a component, or with a phenomenon, and the actual value of that parameter.

safety margin (in connection with the results of analyses). The difference, in physical units, between a threshold that characterizes an acceptance criterion and the result provided by either a best estimate or a conservative calculation. In the case of best estimate calculation, the uncertainty band must be used when defining the safety margin.

scaling. Based upon suitable physical principles, the establishment, in system thermohydraulics, of a correlation between phenomena expected in a nuclear power plant and phenomena measured in smaller scale facilities, or phenomena predicted by numerical tools qualified against experiments performed in small scale facilities. (The term 'nuclear power plant' can be replaced here by 'large scale integral test facility'.)

sensitivity analysis (in the context of uncertainty analysis). Quantification of the degree of impact of the uncertainty from the individual input parameters of the model on the overall model outcome (uncertainty importance analysis).

separate effects test facility (SETF). Experimental loop designed in accordance with a proper set of scaling laws aimed at the simulation of a single phenomenon, or of a restricted number of phenomena, or of the behaviour of a single geometrical zone or of a restricted number of zones. The phenomena are expected to occur in nuclear power plants transient scenarios, and the zones are part of the plant. Phenomena measured in SETFs are expected to be as similar as possible to phenomena expected in the reference plant. (The design of an SETF implies the existence of a reference plant that has already been built or designed.)

sources of uncertainty. Parameters that affect the results of a calculation. These can be part of the input deck (nodalization), including boundary and initial conditions, or can be embedded in the code, including imperfections in physical models, structure and/or inadequacies of the balance equations and of the numerical solution methods.

uncertainty. Measure of scatter in experimental data or calculated values. It is expressed by an interval around the true mean of a parameter resulting from the inability to either measure or calculate the true value of that parameter (scatter). The uncertainty is often given as a (e.g. 95%) probability limit or probability interval.

uncertainty method. Procedures that allow the quantification of the error in code calculation (i.e. the error affecting the code output quantities), taking the various sources of uncertainty into consideration.

uncertainty range (deterministic and probabilistic). Depending on the uncertainty method used, the state of knowledge about an uncertain parameter is given as a 'bounding' range, 'reasonable' uncertainty range or as a probability distribution.

user effect. A user effect is the difference between two sets of calculation results obtained by two code users (or two groups of code users) who use the same code and have access to the same information for setting up the nodalization and for determining the needed input and boundary condition values.

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Deterministic safety analysis is an important tool for confirming the adequacy and efficiency of provisions within the defence in depth concept for the safety of nuclear power plants. IAEA Safety Standards Series No. NS-R-1.2 and Safety Reports Series No. 23 recommend, as one of the options for demonstrating the inclusion of adequate safety margins, the use of best estimate computer codes with realistic input data in combination with the evaluation of uncertainties in the calculation results. The evaluation of uncertainties is an issue of considerable complexity, and this safety report has been developed to complement the existing publications. It provides more detailed information on the methods available for the evaluation of uncertainties in deterministic safety analysis of nuclear power plants and practical guidance in the use of these methods.

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