Science and Technology of Nuclear Installations

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Guest Editors: Cesare Frepoli and Alessandro Petruzzi

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Contents

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Editorial

Scaling, Uncertainty, and 3D Coupled Code Calculations in Nuclear Technology

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System codes such as RELAP, TRACE, CATHARE, or ATHLET are currently used by designer/vendors of NPPs, by utilities, licensing authorities, research organizations including universities, nuclear fuel companies, and by technical supporting organizations. The objectives of using the codes may be quite different, ranging from design or safety assessment to simply understanding the transient behavior of a simple system. However, the application of a selected code must be proven to be adequate to the performed analysis. Thus considerable research efforts have been spent in the last three decades, and as a consequence a wide range of activities has recently been completed in the area of system thermal-hydraulics. Problems have been addressed, solutions to which have been at least partly agreed upon on international ground. These include the need for best-estimate system codes, the general code qualification process, the proposal for nodalization qualification as well as attempts aiming at qualitative and quantitative accuracy evaluations. Moreover, complex uncertainty methods have been proposed, following a pioneering study which attempted, among other things, to account for user effects on code results.

Based on the above considerations, this special issue mostly focuses on the development and application of best-estimate codes emphasizing the role of the *scaling*, *best estimate and uncertainty*, and *3D coupled code calculations* analyses.

In general terms, *scaling* indicates the need for the process of transferring information from a model to a prototype. In system thermal hydraulics, a scaling process, based upon suitable physical principles, aims at establishing a correlation between phenomena expected in an NPP-transient scenario as (a) phenomena measured in smaller scale facilities, or (b) phenomena predicted by numerical tools qualified against experiments performed in small-scale facilities. In connection with this point, owing to limitations of the equations at the basis of system codes, the scaling issue may constitute an important source of uncertainties in code applications.

By definition, a *best-estimate* analysis (the term "best-estimate" is usually used as a substitute for "realistic") is an accident analysis which is free of deliberate pessimism regarding selected acceptance criteria, and is characterized by applying best-estimate codes along with nominal plant data and best-estimate initial and boundary conditions. However, notwithstanding the important achievements and progress made in recent years, the predictions of the best-estimate system codes are not exact but remain uncertain because (a) the assessment process depends upon data almost always measured in small-scale facilities and not in the full-power reactors; (b) the models and the solution methods in the codes are approximate. In some cases, fundamental laws of physics are not considered. Consequently, the results of the code calculations may not be applicable to give exact information on the behavior of a nuclear power plant (NPP) during postulated accident scenarios. Therefore, best-estimate predictions of NPP scenarios must be supplemented by proper uncertainty evaluations in order to be meaningful. The term "*best-estimate plus uncertainty*" was coined for indicating an accident analysis which (1) is free of deliberate pessimism regarding selected acceptance criteria, (2) uses a BE code, and (3) includes uncertainty analysis. Thus, the word "uncertainty" and the need for uncertainty evaluation are strictly connected with the use of BE codes.

Nowadays, advanced *3D coupled neutron-kinetics/thermal-hydraulics* computer tools along with powerful computers can perform realistic best-estimate analyses of complex power plant transients. The interaction between thermal-hydraulics and neutron kinetics is relevant for both the safety and the design of existing nuclear reactors. The results from the application of coupled computational tools provide new insights into the conservatisms for the specification of relevant operational safety margins and can imply new optimizations of emergency operating procedures in existing plants. They also improve knowledge of the physical phenomena in nuclear water reactor technology and can specifically shed light on the interaction between thermal-hydraulics and neutron kinetics that still can challenge the design and the operation of nuclear power plants.

This special issue collects selected lectures delivered at the 3D S.UN.COP (Scaling, Uncertainty, and 3D COuPled code calculations) seminars-trainings whose aim is to transfer competence, knowledge, and experience from about 30 recognized international experts coming from more than 10 different countries and institutions to analysts with a suitable background in nuclear technology. The program of the 3D S.UN.COP offers each year about 60 presentations and 100 hours of parallel code hands-on training subdivided in three weeks and covering the following topics: (a) system codes: evaluation, application, modeling and scaling; (b) international standard problems; (c) best-estimate in system code applications and uncertainty evaluation; (d) qualification procedures; (e) methods for sensitivity and uncertainty analysis; (f) relevant topics in best-estimate licensing approach; (g) industrial applications of the best-estimate-plus-uncertainty methodology; (h) coupling methodologies and applications; (i) computational fluid dynamics codes. From the other side, the parallel hands-on training sessions on numerical codes (such as CATHARE, CATHENA, RELAP5, TRACE, and PARCS) allow the participants to achieve the capability to set up, run, and evaluate the results of a numerical tool through the application of the proposed qualitative and quantitative accuracy evaluation procedures. Finally, the 3D S.UN.COP seminars provides a forum for exchanges of ideas through scientific presentations and dialogue among representatives of the worlds of academy, research laboratories, industry, regulatory authorities, and international institutions.

In the first paper, A. Petruzzi et al. emphasized the role of the computer code user that represents one of the main sources of uncertainty influencing the results of system code calculations. This influence is commonly known as the "user effect" and stems from the limitations embedded in the codes as well as from the limited capability of the analysts to use the codes. The paper describes a systematic approach to training code users who, upon completion of the training, should be able to perform calculations making the best possible use of the capabilities of best-estimate codes.

In the second paper, A. Petruzzi, and F. D'Auria presented the commonly used system thermal-hydraulic codes such as RELAP, TRACE, CATHARE, or ATHLET for reactor-transient simulations. Whereas the first system codes, developed at the beginning of the 1970s, utilized the homogenous equilibrium model with three balance equations to describe the two-phase flow, nowadays the more advanced system codes are based on the so-called "two-fluid model" with separation of the water and vapour phases, resulting in systems with at least six balance equations. However, notwithstanding the huge amounts of financial and human resources invested, the results predicted by the code are still affected by errors whose origins can be attributed to several reasons as model deficiencies, approximations in the numerical solution, nodalization effects, and imperfect knowledge of boundary and initial conditions. In this context, the existence of qualified procedures for a consistent application of qualified thermal-hydraulic system code is necessary and implies the drawing up of specific criteria through which the code-user, the nodalization, and finally the transient results are qualified.

In "International Standard Problems and Small Break Loss-Of-Coolant Accident (SBLOCA)," N. Aksan considered five small break LOCA-related ISPs since these were used for the assessment of the advanced best-estimate codes. The considered ISPs deal with the phenomenon typical of small break LOCAs in Western design PWRs. The experiments in four integral test facilities, LOBI, SPES, BETHSY, ROSA IV/LSTF, and in the recorded data during a steam generator tube rupture transient in the DOEL-2 PWR (Belgium) were the basis of ISP calculations. The statistical evaluation of the general data obtained from these ISPs is summarized. Some lessons learned from these small break LOCA ISPs are identified in relation to code deficiencies and capabilities, progress in the code capabilities, possibility of scaling, and various additional aspects. ISPs are providing unique material and benefits for some safety related issues. Some of the technical findings and benefits provided by small break LOCA ISPs are provided as conclusions and recommendations.

In the next paper, A. Petruzzi, and F. D'Auria presented the evaluation of uncertainty methodologies as necessary supplement of best-estimate calculations performed to understand accident scenarios in water-cooled nuclear reactors. The needs come from the imperfection of computational tools, on the one side, and the interest in using such a tool to get more precise evaluation of safety margins. The paper reviews the salient features of two independent approaches for estimating uncertainties associated with predictions of complex system codes. Namely, the propagations of code input error and calculation output error constitute the keywords for identifying the methods of current interest for industrial applications. Throughout the developed methods, uncertainty bands can be derived (both upper and lower) for any desired quantity of the transient of interest. For one case, the uncertainty method is coupled with the thermal-hydraulic code to get the code with capability of internal assessment of uncertainty, whose features are discussed in more detail.

The task of regulatory body staff reviewing and assessing a realistic large break loss-of-coolant accident evaluation model is discussed by R. Galetti in the next paper facing the actual regulatory licensing environment related to the acceptance of the analysis of emergency core cooling system performance. Especially, focus is directed to the question of how to fulfill the requirement of quantifying the uncertainty in the calculated results when they are compared to the acceptance criteria for this system. When using a realistic evaluation model to analyze the loss-of-coolant accident, different approaches have been used in the licensing arena. The Brazilian regulatory body has concluded that, in the current environment, the independent regulatory calculation is recognized as a relevant support for the staff decision within the licensing framework of a realistic analysis.

In the sixth paper, H. Glaeser summarized the basic techniques of the GRS uncertainty method together with applications to a large break loss-of-coolant accident on a reference reactor as well as on an experiment simulating containment behavior. A significant advantage of this methodology is that no a priori reduction in the number of uncertain input parameters by expert judgement or screening calculations is necessary to limit the calculation effort. All potentially important parameters may be included and the number of calculations needed is independent of the number of uncertain parameters accounted for in the analysis. A challenge in performing uncertainty analyses with the GRS methodology is the specification of ranges and probability distributions of input parameters.

C. Frepoli presented the paper entitled "An Overview of Westinghouse Realistic Large Break LOCA Evaluation Model." Since the 1988 amendment of the 10 CFR 50.46 rule in 1988, Westinghouse has been developing and applying realistic or best-estimate methods to perform LOCA safety analyses. Westinghouse methodology is based on the use of the WCOBRA/TRAC thermal-hydraulic code. The paper starts with an overview of the regulations and its interpretation in the context of realistic analysis. The CSAU (code scaling, applicability, and uncertainty) roadmap is reviewed in the context of its implementation in the Westinghouse evaluation model. An overview of the code (WCOBRA/TRAC) and methodology is provided. Finally, the recent evolution to nonparametric statistics in the current edition of the Westinghouse methodology is discussed. Sample results of a typical large break LOCA analysis for a PWR are provided.

The next paper by R. Martin and L. O'Dell illustrates the development considerations of AREVA NP Inc.'s realistic LBLOCA analysis methodology. The AREVA NP RLBLOCA methodology is a CSAU-based methodology for performing best-estimate large-break LOCA analysis. The methodology addresses all of the expressed steps of the CSAU process. The key challenge to this process has been the defense of declared engineering judgment and the demonstration of the methodologies' range of applicability. This was accomplished by careful characterization of dominant LOCA parameters and emphasis on validation through sensitivity studies and the statistical nature of the methodology. The generic AREVA NP RLBLOCA methodology was approved by the USNRC in April 2003 and is now being applied to several nuclear power plants serviced by AREVA NP Inc.

In the next paper, D. Novog and P. Sermer provided a novel and robust methodology for determination of nuclear reactor trip set points, which accounts for uncertainties in input parameters and models, and for the variations in operating states that periodically occur. The paper presents the general concept used to determine the actuation set points considering the uncertainties and changes in initial conditions, and allowing for safety system instrumentation redundancy. The results demonstrate unique statistical behavior with respect to both fuel and instrumentation uncertainties, which has not previously been investigated.

F. Reventos et al. illustrated the usefulness of computational analysis for operational support in the paper before the last. In the first part, he described the specific aspects of thermal-hydraulic analysis tasks related to operation and control and, in the second part, they briefly presented the results of three examples of performed analyses. All the presented examples are related to actual situations in which the scenarios were studied by analysts using thermal-hydraulic codes and prepared nodalizations. The paper also includes a qualitative evaluation of the benefits obtained through thermal-hydraulic analyses aiming at supporting operation and plant control.

In the last paper, H. Ikeda et al. reviewed activities relevant to the boiling water reactor (BWR) stability phenomenon, which has a coupled neutronic and thermal-hydraulic nature, from the viewpoint of model and code developments. Industrial organizations have developed and improved the BWR stability analysis using computational tools specific for the reduced-order frequency-domain and three-dimensional time-domain codes. The first category is currently applied to the BWR stability design analysis, while the latter has been exploited to understand the complicated phenomena related to BWR stability. Proposals to apply best-estimate analysis code with the statistical safety evaluation methodology are currently under study. This will allow better evaluation of the stability exclusion region, and will be consequently applied to the BWR plants with the extended core power uprate.

We believe that the collection of papers in this special issue illustrates the great variety of topics and problems in the nuclear technology for which advanced tools are available and applicable.

Finally, we would like to take the opportunity to express our thanks to all authors who have submitted papers to this special issue and to our colleagues who devoted their valuable time reviewing these manuscripts.

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Research Article

Regulatory Scenario for the Acceptance of Uncertainty Analysis Methodologies for the LB-LOCA and the Brazilian Approach

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The task of regulatory body staff reviewing and assessing a realistic large break loss-of-coolant accident evaluation model is discussed, facing the actual regulatory licensing environment related to the acceptance of the analysis of emergency core cooling system performance. Especially, focus is directed to the question of how to fulfill the requirement of quantifying the uncertainty in the calculated results when they are compared to the acceptance criteria for this system. As it is recognized that the regulation governing the loss-of-coolant accident analyses was originally developed by the United States Nuclear Regulatory Commission, a description of its evolution is presented. When using a realistic evaluation model to analyze the loss-of-coolant accident, different approaches have been used in the licensing arena. The Brazilian regulatory body has concluded that, in the current environment, the independent regulatory calculation is recognized as a relevant support for the staff decision within the licensing framework of a realistic analysis.

1. INTRODUCTION

The objective of this paper is to discuss the regulatory licensing environment related to the acceptance of the analysis of emergency core cooling system (ECCS) performance in light water reactors when using a realistic or best-estimate evaluation model. The focus is directed to the question of how to meet the requirement of quantifying the uncertainty in the calculated results when they are compared to the acceptance criteria for this system.

It also included the experience of the Brazilian nuclear regulatory body (CNEN) reviewing and assessing the Angra 2 nuclear power plant (NPP) large-break loss-of-coolant accident (LB-LOCA) analysis, submitted for licensing with a realistic evaluation methodology.

2. REGULATING THE USE OF BE + U

The United States Nuclear Regulatory Commission (US-NRC) emergency core cooling systems acceptance criteria, issued in 1974 [1], is recognized as a highly conservative approach due to limitations in knowledge at that time. This relevant aspect was identified and dealt with by the nuclear community through a huge effort in the reactor-safety research area. For additional details, see [2–6].

In 1983, based on experimental programs results, the ability of advanced computer codes to predict the behavior during an LOCA was demonstrated, and the conservatism in Appendix K could be quantitatively estimated. Because of this, through the release of SECY-83-472 [7], the NRC adopted an interim approach for evaluation models retaining the features of Appendix K which were recognized as requirements but allowing the use of best estimate methods in models and correlations. Even still conservative, this approach was the first step on licensing decision making based on realistic calculations.

On September 16, 1988, the NRC amended the requirements of 10 CFR 50.46 [8] reflecting the improved understanding of the thermal-hydraulic phenomena occurring during the loss-of-coolant accidents, obtained by the results of extensive research programs sponsored by the NRC and the nuclear industry. In Brazil, CNEN adopted this revision which allows, as an option, the use of realistic evaluation models to calculate the performance of the emergency core cooling system. In such cases, the LOCA analysis will fulfill the requirement of identifying and evaluating the uncertainty in the analysis methods and inputs, and this uncertainty must be considered when comparing the calculated results with the acceptance criteria so that there is a high probability that the criteria will not be exceeded.

This revision of 10 CFR 50.46 allows licensees or applicants to use either the conservative evaluation model defined in Appendix K, with its conservative analysis methods, or a realistic evaluation model (best-estimate plus uncertainty analysis methods). The Regulatory Guide 1.157 [9] describes acceptable models, correlations, data, model evaluation procedures, and methods for meeting the specific requirements for a realistic calculation of ECCS performance during a LOCA.

Despite of that, there is still a lack of an established set of specific regulatory requirements and guidance applied to the acceptance of the uncertainty calculation related to the results of a realistic evaluation model used to analyze the LOCA. On January 11, 2001, the Advisory Committee on Reactor Safeguard (ACRS) of USNRC addressed the question of how the perceived weaknesses of the thermal-hydraulic codes may affect the regulatory role, and already emphasized in a Letter Report [10], "We perceive a need for the staff to be more specific about what are acceptable methods of deriving and expressing the uncertainties in codes and how these methods are to be used in the regulatory context".

More recently, NRC has issued section 15.0.2 of the Standard Review Plan [11] describing the review process and acceptance criteria for analytical models and computer codes used to analyze the accident and transient behavior, including methods to estimate the uncertainty in best-estimate LOCA calculation. Additionally, guidance to the industry was issued, set forth in Regulatory Guide 1.203 [12]. Despite of that, as it has been pointed out by ACRS in its January 11, 2001 Letter Report related to Regulatory Guide 1.157, these new regulatory guidance documents remain very qualitative and leaves considerable latitude in interpretation.

In parallel, NRC has been conducted research, together with industry, related to the acceptance criteria for ECCS. As an example, it should be mentioned that the ongoing development of a performance-based option for the embrittlement criteria in 50.46(b) [13–15], and also the proposed rule for a voluntary alternative to 10 CFR 50.46, related to the definition of LOCA break sizes [16].

In the United States, the first NRC approved best-estimate LOCA methodology was the Westinghouse methodology [17], patterned after the Code Scaling, Applicability, and Uncertainty evaluation methodology (CSAU), and uses response surfaces to estimate PCT uncertainty distribution with the 95th percentile PCT determined from a Monte Carlo sampling and accepted as the licensing basis PCT. In 1999, it was extended to other plants design (AP600 and 2-loops plants with upper plenum injection). By 2000, 14 plants in the United States had Westinghouse BELOCA methodology as a licensing basis and it was also used for Ringhals unit 2 in Sweden [18].

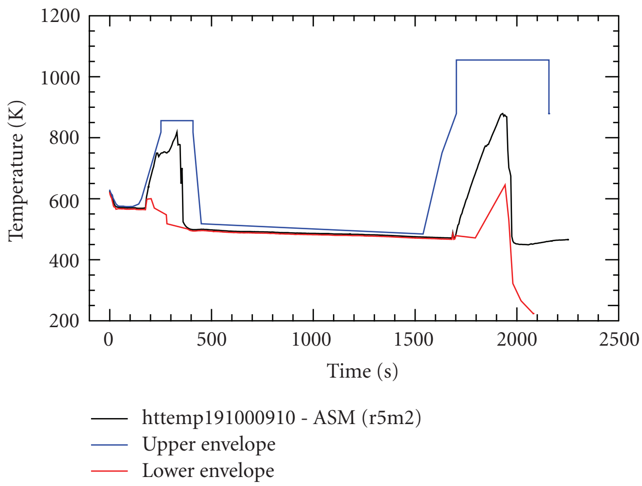

Figure 1: JONATER application of UMAE to Angra 1 NPP SB-LOCA: uncertainty bands for the PCT.

Framatome ANP has submitted its realistic LB-LOCA methodology and got NRC approval in 2003 [19]. It follows CSAU approach but was the first to use a nonparametric order statistic method, eliminating the need for response surfaces. By 2006, there were seven completed realistic LB-LOCA analyses with 3-loop and 4-loop Westinghouse and Combustion Engineering pressurized water reactors [20].

By 2004, Westinghouse updated its methodology to use nonparametric order statistic, and an advanced statistical treatment of uncertainty method (ASTRUM) [21] was approved for licensing. In US, by 2006, 24 plants are licensed or analyzed with Westinghouse 1996 and 1999 BELOCA methodologies and 10 plants are analyzed or being analyzed with ASTRUM [18].

It is worthwhile to mention the ongoing issue at the regulatory arena with the use of nonparametric order statistics methodology to demonstrate that the criteria in 10 CFR 50.46(b) are satisfied. The number of ECCS performance-evaluation model runs accepted to demonstrate a probability that the criteria will not be exceeded is different in two similar realistic LB-LOCA methodologies approved by the NRC [19, 21]. Such difference is due to the approach used to demonstrate the simultaneous satisfaction of the first three criteria in 10 CFR 50.46(b), peak cladding temperature, peak local oxidation, and corewide oxidation. There are still undergoing discussions on this philosophical issue [22–26].

In Germany, the use of best-estimate codes is allowed, in combination with conservative initial and boundary conditions, and efforts are being conducted to include uncertainty evaluation in the regulation with a revision in the German nuclear regulation. There is also a recommendation of the Reactor Safety Commission to perform LOCA licensing analysis [27].

In Canada, the Canadian Nuclear Safety Commission recently conducted a research program that resulted in a guide for safety assessment and applications of best-estimate analysis and uncertainty methodology [28].

3. BRAZILIAN REGULATORY EXPERIENCE

Angra 2 NPP is a 4-loop Siemens design 1300 MWe pressurized water reactor that started commercial operation in 2001. The best-estimate LOCA approach was formally adopted by the utility Eletronuclear (ETN) in 1994. By 1998, when the realistic LB-LOCA analysis was submitted, based on CSAU methodology to evaluate the uncertainty, there were only few applications of realistic evaluation models in the licensing arena.

Aiming at performing a consistent safety review and assessment of this analysis, the Brazilian regulatory body trained its staff and relied upon two international consultants, the German institute GRS (Gesellschaft fur Anlagen und Reaktorsicherheit) and the University of Pisa.

The cooperation with many international institutions involved in accident-analysis research provided a relevant technical background for the regulatory staff. In the same time, the national thermal-hydraulic journey (JONATER), coordinated by CNEN, has promoted the integration of seven institutions (regulatory body, research institutes, and utility) of the Brazilian nuclear sector. One result of JONATER was the use of an uncertainty methodology applied as an exercise for Angra 1 NPP, a Westinghouse 630 MWe 2-loop pressurized water reactor. The uncertainty bands were estimated with the UMAE [29] method for the results of the small-break LOCA simulated with the Relap5Mod2 code, as it is shown in Figure 1 [30].

UMAE is an uncertainty methodology based on accuracy extrapolation resulting from a comparison between code results and relevant experimental data obtained in experimental facilities. These facilities were simulated, for the chosen transient scenario, with an established nodalization that will be the basis for the nodalization adopted in the plant calculation. The extrapolated accuracy is superimposed directly to the results of the plant calculation. Uncertainty bands are constituted by a set of "punctual" error bands in the x-y plane (where x is the time t and y is Y_C quantity). Each value Y_C at a time t can be characterized by an error DY_C in the "y" direction and by an error Dt in the "x" direction. The total uncertainty is the superimposition of these two errors.

As the estimation of Angra 1 small-break LOCA uncertainty bands was an exercise for the application of an uncertainty methodology, for the accuracy calculation, only the large scale test facility (LSTF) database was considered (experimental and Relap5/Mod2 results for the SB-CL-21 test). It is important to mention that the accuracy should be obtained from more tests to avoid some poor accuracy that eventually can result for some specific parameter. For instance, code simulation of the LSTF experiment yielded a result for the heater rod temperature and time of its occurrence far from the verified experimental value. Therefore, the lower uncertainty band at the end of the transient for the peak cladding temperature shows no physical results due to the limited number of experimental data used.

The Angra 2 LB-LOCA analysis presented in the final safety-analysis report was reviewed by CNEN staff taking into account the two independent reviews performed by the international consultants. As a result, a preliminary safety-evaluation report (SER) requested additional information (RAI), with a total of 27 questions to the applicant, each one is classified according to their significance to safety [31].

Table 1 lists the main steps in the review and assessment process of Angra 2 NPP LB-LOCA analysis.

The Siemens uncertainty methodology applied to Angra 2 followed, essentially, the CSAU approach (Phenomena Identification Ranking Table, code capabilities for accident scenario) and used Monte Carlo calculations with response surface. The treatment of the uncertainties is performed separately from three basic categories: code uncertainties (statistical quantification of difference between calculated and measured PCT), plant parameters uncertainties (statistical variations), and fuel parameters uncertainties (statistical variations). Some additional parameters related to uncertainties have been required to be run at combined worst-case conditions. These parameters are break area and location, axial core power distribution, worst-case single failure and repair assumption, loss of offsite power, and reactor kinetics.

This uncertainty analysis is such that the 95% probability PCT was generated by using Monte Carlo to combine uncertainties from the three sources. The two other criteria (maximum cladding oxidation and hydrogen generation) were calculated considering conservative assumptions.

The number of data points, used to determine code accuracy through the quantification of the differences between calculated and measured results for LOFT and CCTF experiments, was one example of RAI from the preliminary SER. It was further required from the applicant to verify the implications of considering additional relevant experimental data into code integral uncertainties. Additionally, the applicant presented code uncertainty quantification with more experimental data.

After the issuance of the preliminary SER, the importance of an independent regulatory calculation was recognized. Together with CNEN staff, the University of Pisa performed independent calculation [32, 33]. Based on its conclusions, three requests for additional information were issued to the applicant, mainly related to plant modelling, which has to be consistent with those used for the validation calculations.

As future applications, the Brazilian regulatory body has already been informed by the utility ETN of its intention to uprate 6% the Angra 2 power together with a change in the fuel design, replacing it to a high thermal performance fuel with M5 fuel cladding. This will require the reanalysis of the LB-LOCA with uncertainty quantification.

Furthermore, for Angra 1 NPP steam-generators replacement, the utility will submit a realistic evaluation model for the LB-LOCA, using the Westinghouse methodology that encompasses the WCOBRA/TRAC code with the ASTRUM methodology for uncertainty calculation. Additionally, the power will be uprated 5% and a new fuel design will be used (16 next-generation fuel, developed jointly by Westinghouse, Korea Nuclear Fuel (KNFC), and Indústrias Nucleares do Brasil (INB)).

Table 1: Angra 2 NPP LB-LOCA Review.

Activity	Date
Submission of FSAR with the LOCA analysis	1998
CNEN's Preliminary Safety Assessment	1998
GRS Expert Mission-FSAR LBLOCA Analysis and Uncertainty Method revision	1998
IAEA Expert Mission/Pisa Uni-FSAR LBLOCA Analysis and Uncertainty Method revision	1999
Issuance of CNEN Preliminary SER with 27 RAI	1999
GRS Expert Follow-up Mission	1999
Answers to the request of additional information	1999
Licensing meeting: Utility, Siemens, CNEN, GRS and Pisa University	1999
Submission of Generic Thermal Mechanics Analysis of Fuel Failures after a LOCA	1999
Submission of Angra 2 Specific Thermal Mechanics Analysis of Fuel Failures after a LOCA	1999
Emission of CNEN SER about ECCS Technical Specification	2000
Pisa University Technical Consultancy to CNEN-Regulatory audit analysis	2001
Issuance of CNEN SER: additional RAI on core nodalization and uncertainty quantification	2001
Independent LB-LOCA calculation used to check the request of a temporary 6% power uprate	2002

4. REGULATORY INDEPENDENT ANGRA 2 LB-LOCA ANALYSIS DESCRIPTION

The independent calculation included the LB-LOCA calculation with Relap5/Mod3.2.2 Gamma code and the uncertainty evaluation with the CIAU method (code with capability of internal assessment of uncertainty) [34].

In this application, the CIAU method used UMAE methodology for uncertainty quantification that is based upon propagation of code output error and does not rely on statistics. The inaccuracies are obtained by experimental calculation comparison and are extrapolated to get uncertainty. The database for accuracy extrapolation was derived from 32 experimental transients that were calculated by Pisa University with Relap5/Mod3.2.2 Gamma code.

The independent LB-LOCA calculation activities were planned with the objective to consider the steps presented in a best-estimate analysis: a qualified nodalization development (steady-state level and on-transient level), transient reference-case calculation, uncertainty evaluation, and comparison between the results obtained in the sensitivity studies and in the uncertainty analysis.

A "fictitious" 3D nodalization of the reactor pressure vessel was adopted considering the experience in the analysis of the upper plenum test facility experiments [35]. Two main nodalizations were established at the beginning of the studies, characterized by:

(i) nonuniform upper plenum behavior, pursuing the nodalization strategy of the utility ETN in the FSAR analysis, top-down flow allowed only in the determined breakthrough channels [36];

(ii) uniform upper plenum behavior with top-down flow allowed in all channels except in the hot assembly, with the worst conditions for core cooling inside the hydraulic hot assembly, by "hydraulically separating" the hot fuel assembly from the average core region.

After defining a reference calculation and performing the sensitivity study, the reference-case nodalization chosen was the one without cross-flow simulation between the hot fuel assembly and the rest of the core (denominated tr12), that might bring undue conservatism in the results. The one considering this cross flow (denominated a2n04x) could be the reference case if experimental data was available to establish the flow energy-loss coefficients. Therefore, for the a2n04x run, these coefficients were established through engineering judgment without an experimental basis. The use of S-RELAP5 code in the Angra 2 FSAR LB-LOCA analysis considers implicit this cross flow through the full two-dimensional treatment added to the hydrodynamic field equations.

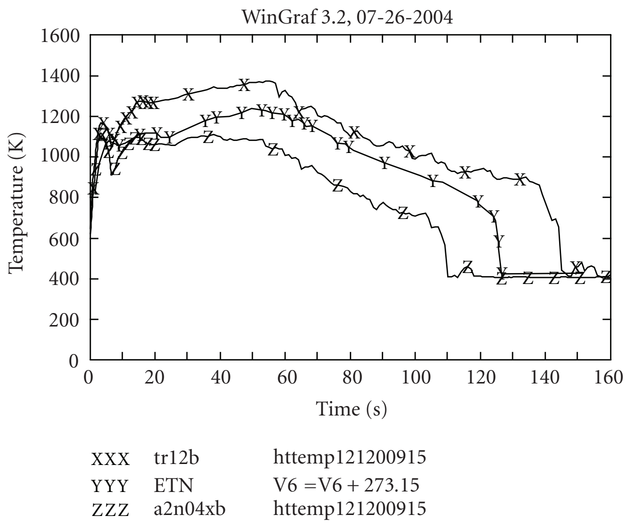

Figure 2: Cladding temperature of the hot rod.

Figure 2 shows a comparison of the reference calculation result to FSAR result for the peak cladding temperature (PCT) for the "base case". In the FSAR analysis, this "base case" is defined in the adopted ETN methodology as the nominal condition for the uncertainty analysis. This uncertainty analysis is such that the 95% probability PCT was

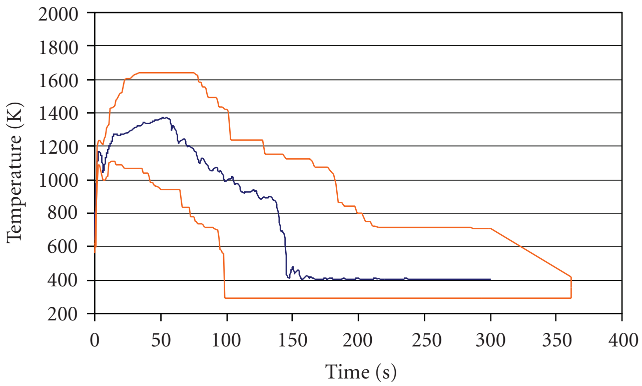

Figure 3: Uncertainty bands for rod surface temperature at 2/3 of the core height-CIAU result.

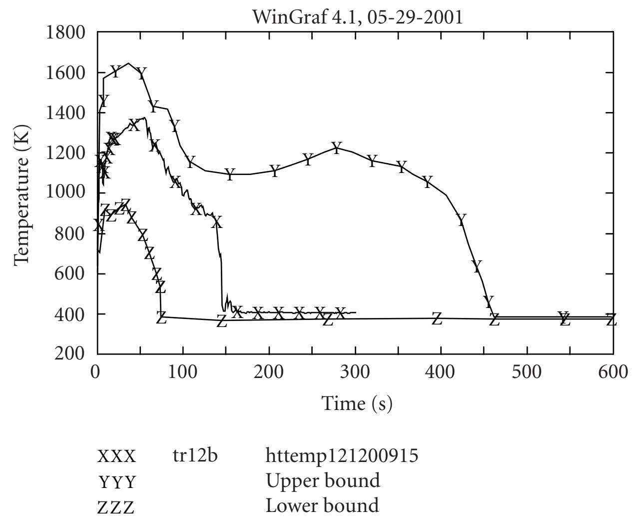

Figure 4: Angra 2 NPP LB-LOCA sensitivity study: upper and lower bounds from the rod surface temperature. Envelope uncertainty evaluation.

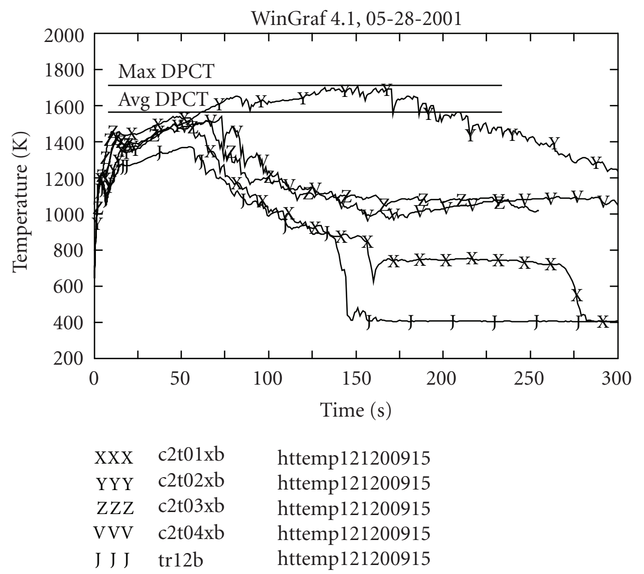

Figure 5: Angra 2 NPP LB-LOCA sensitivity study, achievement of a deterministic value for ΔPCT. Labels XXX through VVV representing code runs based on combination of three variations of input parameters.

generated by using Monte Carlo to combine uncertainties from the three sources. The “base case” is the reference case for the determination of the calculation-design matrix used to generate data for fitting the response surfaces. Also, the “base case” is the reference case where the effects of the plant uncertainties are determined.

The comparison of the PCT from the “base case” and the “reference calculation” indicates a discrepancy, with a higher value observed in independent calculation result. In the case of “reference calculation”, it is shown that the removal of conservatism of assuming no cross flow to the hot channel substantially lowers the reported value. This outcome confirms the importance of assessing, by using experimental data, the cross flow to the hot channel if this is considered.

In the independent regulatory calculation, automatic uncertainty bands for primary-system pressure, mass inventory, and rod surface temperature at 2/3 of the core active height are generated by the CIAU method and constitute the results of the application. Figure 3 shows the result for PCT.

The number of experiments, which were used to derive code uncertainty from CIAU, is limited. Therefore, a sensitivity study has been performed to confirm the results obtained from this methodology. Additional objective was to confirm that the impact of an assigned input parameter upon the results is dependent on the nodalization.

A comprehensive-sensitivity study has been carried out including two series of calculations. Starting from the two main nodalizations, single parameters are varied in each code run. Six groups of input parameters are distinguished: “fuel”, “nodalization”, “loop hydraulics”, “PSA and ECCS”, “neutronics”, and “others”. The number of performed runs was 112.

Thefirstseries aims at confirming the influence of selected input parameters upon the LB-LOCA predicted scenario, and showing the importance of nodalization upon the same prediction when an assigned input parameter is varied. Code runs with single change of input parameters and with realistic variation ranges were used for the envelope uncertainty evaluation. Examples of input parameters varied, at one time, in the code run: fuel (gap thickness, UO2 conductivity, gap conductance), loop hydraulics (critical flow model, spacer grid modelling, reactor pressure-vessel bypass flow), nodalization (upper-plenum pressure drop, counter current-flow limitation in the core), PSA and ECCS (loss of offsite power delay, components actuation), and neutronics (moderator coefficient, decay power). The result is shown in Figure 4 where the envelope of all the considered calculations is reported.

The second series aims at determining boundary values for PCT. Three input parameters, chosen among those considered in the first series of calculations, are selected and varied simultaneously in each run. Examples of chosen parameters are UO2 conductivity, break-discharge coefficient, ECCS components actuation, decay power, and gap conductance. The ranges of variations are maximized. These code runs are adopted for the deterministic evaluation of the uncertainty (see Figure 5).

The parameter ΔPCT is defined as the difference between the PCT of the reference calculation and the PCT obtained from the generic sensitivity run. The dispersion of results for ΔPCT obtained from the first series of code runs provides an overall picture of the influence of nodalization upon predictions, confirming the importance of the nodalization upon the predicted scenario.

The following valuable results were obtained.

(i) The upper and lower uncertainty bands from the envelope uncertainty evaluation in Figure 4 can be compared with the CIAU uncertainty bands in Figure 3. Therefore, the uncertainty results obtained by CIAU are supported by the outcome of the sensitivity study.

(ii) The uncertainty ranges predicted by CIAU, resulting from the sensitivity study and the ones reported in the FSAR, are comparable.

The adopted noding scheme, that is, the nodalization, has been found as the critical issue of the study. The nodalization features affect the prediction of the safety relevant parameters, the interpretation of the performed “sensitivity” runs, and the use of the outcomes from the uncertainty method. Namely, the application of a 1D designed assessed code, having at the basis a fictitious 3D model of the vessel, requires a number of engineering choices. These choices have been proven to impact noticeably the results, and must be adequately supported by a suitable experimental evidence.

Results from a best-estimate code prediction are largely affected by the nodalization features. Therefore, the full demonstration of the nodalization quality at the “steady state” and at the “on-transient” level is needed to derive meaningful conclusions about the safety performance of the concerned NPP. Considering Angra 2 features, basically, the hot leg injection, a decisive importance is revealed by the upper plenum and core outlet modeling.

5. CONCLUSIONS

As described in the previous sections, when using a realistic evaluation model to analyze the LOCA, different approaches have been used in the licensing arena to demonstrate the fulfillment of the ECCS acceptance criteria.

Besides the different approaches, the regulators are aware of the development in the uncertainty methodologies and, therefore, further actions should be required even after a methodology has been accepted.

The Brazilian regulatory body is monitoring these activities and it has concluded that, in the current environment, the independent regulatory calculation is recognized once again as a relevant support for the staff decision within the licensing framework of a realistic LB-LOCA analysis.

In the case of Angra 2 LB-LOCA, the independent calculation complemented, on a quantitative basis, the task of reviewing and assessing, and allowed to check the completeness and consistency of the submitted accident analysis. The use of an uncertainty methodology (CIAU) that has a different approach compared to the designer approach (Siemens) contributed to the understanding of the validity limits of the results submitted by the licensee within the FSAR. Conclusions are provided in relation to the acceptability of the actual safety margins of the Angra 2 NPP.

In the case of Angra 1 LB-LOCA reanalysis for the steam-generators replacement, to be submitted with Westinghouse methodology, the ASTRUM methodology uses a nonparametric order-statistics methodology to demonstrate that the criteria in 10 CFR 50.46(b) are satisfied.

The different approaches observed in the nuclear-power plants in Brazil increase the staff effort to deal with the licensing process. For a small size regulatory body, this diversity of methods, to demonstrate the fulfillment of the ECCS acceptance criteria, indicates a challenge to be faced with technical support organizations providing worldwide recognized experts in the use of best-estimate tools to contribute in the review and assessment process.

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Research Article

Approaches, Relevant Topics, and Internal Method for Uncertainty Evaluation in Predictions of Thermal-Hydraulic System Codes

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The evaluation of uncertainty constitutes the necessary supplement of best-estimate calculations performed to understand accident scenarios in water-cooled nuclear reactors. The needs come from the imperfection of computational tools, on the one side, and the interest in using such a tool to get more precise evaluation of safety margins. The paper reviews the salient features of three independent approaches for estimating uncertainties associated with predictions of complex system codes. Namely, the propagations of code input error and calculation output error constitute the keywords for identifying the methods of current interest for industrial applications, while the adjoint sensitivity-analysis procedure and the global adjoint sensitivity-analysis procedure, extended to performing uncertainty evaluation in conjunction with concepts from data adjustment and assimilation, constitute the innovative approach. Throughout the developed methods, uncertainty bands can be derived (both upper and lower) for any desired quantity of the transient of interest. For one case, the uncertainty method is coupled with the thermal-hydraulic code to get the code with capability of internal assessment of uncertainty, whose features are discussed in more detail.

1. INTRODUCTION

In September, 1988, following the improved understanding of ECCS (emergency core cooling system) performance during various reactor transients, the NRC reviewed and amended the requirements previously fixed in the 10CFR50 (§50.46 and App. K). As a consequence of the conservatism of the methods specified in App. K, being responsible of reactor-operating restrictions and according to industry requests for improved evaluation models, the existing approach was oriented toward basing the licensing decisions on realistic or so-called best-estimate (BE) calculations of plant behavior. The amendments essentially concern the permission of using BE models for ECC performance calculation as an alternative to codes (evaluation models) that use the App. K conservative requirements. The rule changes also include the quantification of uncertainties of best-estimate calculation.

Notwithstanding the efforts made for qualifying the codes and the feedback upon the development, results of system thermal-hydraulics predictions are still affected by noticeable errors, including the unavoidable approximations in the constitutive equations, from the limited capabilities of numerical solution methods, from uncertainties in the knowledge of boundary and initial conditions, and from errors in setting up the nodalization. As a consequence, the result of a best-estimate prediction by a system thermal-hydraulics code, not supplemented by the proper uncertainty evaluation, constitutes nonsense.

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

The first framework for calculating the uncertainty was proposed by US NRC and denominated code scaling, applicability, and uncertainty (CSAU, see [4]). The application of the CSAU methodology resulted in the calculation of the peak-cladding temperature (PCT) during an LBLOCA design basis accident event for a Westinghouse 4-loop pressurized water reactor with the uncertainty to a 95% confidence level. The peak temperature was calculated using the TRAC thermal-hydraulic analysis code and was given as a single-valued number with uncertainty bands. In the meantime, a number of uncertainty methodologies were proposed in other countries. 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. Presently, uncertainty bands can be derived (both upper and lower) for any desired quantity throughout the transient of interest, not only point values like peak cladding temperature.

This paper reviews the salient features of three independent approaches for estimating uncertainties associated with predictions of complex system codes. The origins of the problem and topics relevant for uncertainty evaluation (called hereafter TRUE) are discussed with the aim to analyze how they have been identified and characterized in each method. Finally, a methodology based on internal assessment of uncertainty is presented as an answer to the identified limitations.

2. SOURCES OF UNCERTAINTY IN THERMAL-HYDRAULIC SYSTEM CODES

A fundamental step in the application of best-estimate (BE) method to the safety analysis of nuclear-power plants (NPP) is the identification and characterization of uncertainties. This is connected with the approximate nature of the codes and of the process of code applications. In other words, "sources of uncertainty" affect predictions by BE codes and must be taken into account. The major sources of uncertainty in the area of safety analysis are represented by the:

(i) code or model uncertainty (associated with the code models and correlations, solution scheme, model options, data libraries, deficiencies of the code, and simplifying assumptions and approximations);
(ii) representation or simulation uncertainties (accuracy of the complex facility geometry, 3D effects, control, and system simplifications) including the scaling issue;
(iii) plant data uncertainties (unavailability of some plant parameters, instrument errors, and uncertainty in instrument response).

In addition, the so-called "user effect" is implicitly present and characterizes each of the broad classes of uncertainty above mentioned.

A more detailed list of uncertainty sources, some of them supported by documented evidences (see Figures 1–6), is given hereafter, where an attempt has been made to distinguish "independent" sources of "basic" uncertainty. Complex interactions among the basic uncertainty sources are expected and justify (in advance) the complex structure of an uncertainty method. Comprehensive research programs have been completed [5] or are, in progress [6, 7], aimed at thermal-hydraulic system code assessment and improvement to reduce the influence of the basic uncertainties upon the results.

(A) Balance (or conservation) equations are approximate, that is,

(i) all the interactions between steam and liquid are not included,
(ii) The equations are solved within cylindrical pipes without consideration of geometric discontinuities (situation not common for code applications to the analysis of NPPs transient scenarios).

(B) Presence of different fields of the same phase, for example, liquid droplets and film. However the codes consider only one velocity per phase and this results in an another source of uncertainty.

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

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

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

(F) The 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, that is, transformation of kinetic energy into heat. The amount of the transformation of energy is not necessarily within the capabilities of current codes, thus constituting a further specific energy source.

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

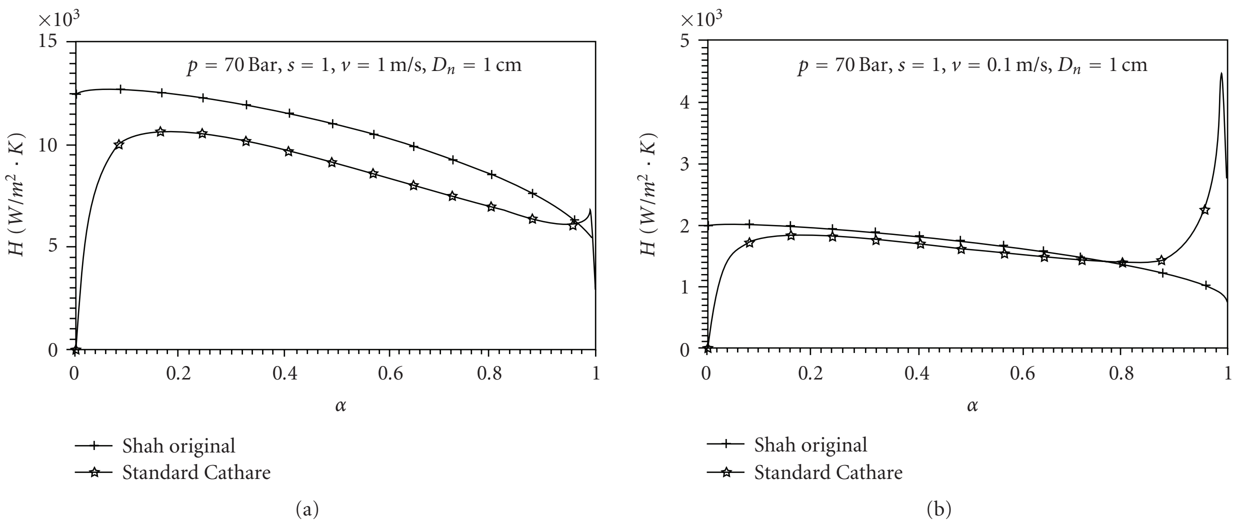

Figure 1: Comparison between results of the original Shah correlation for condensation heat transfer coefficient and Shah correlation after implementation in the Cathare code (not latest version)-two different steam velocities.

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

(i) The ranges of validity are not fully specified. For instance, pressure and flow rate ranges are assigned, but void fraction or velocity (or slip ratio) ranges may not be specified.

(ii) Relationships are used outside their range of validation. Once implemented into the code, the correlations are applied to situations, where, for instance, geometric dimensions are different from the dimensions of the test facilities at the basis of the derivation of the correlation. One example is given by the wall-to-fluid friction in the piping connected with reactor pressure vessel: no facility has been used to derive (or to qualify) friction factors in two-phase conditions when pipe diameters are of the order of one meter. In addition, once the correlations are implemented into the code, no (automatic) action is taken to check whether the boundaries of validity, that is, the assigned ones, are overpassed during a specific application.

(iii) Correlations are implemented approximately into the code. The correlations, apart from special cases, are derived by scientists or in laboratories that are not necessarily aware of the characteristics or of the structure of the system code where the correlations are implemented. Furthermore, unacceptable numeric discontinuities may be part of the original correlation structure. Thus correlations are "manipulated" (e.g., extrapolated in some cases) by code developers with consequences not always ascertained. Figure 1 shows how a valid and qualified correlation (Shah correlations, at two different velocities, for the condensation heat transfer) has been (necessarily) implemented into a system code.

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

(I) A paradoxwill be noted: a "steady-state" and "fully developed" flow condition is a necessary prerequisite or condition adopted when deriving correlations. In other terms, all qualified correlations must be derived under the steady state and fully developed flow conditions. However, almost in no region of the NPP, those conditions apply during the course of an accident.

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

(K) The code user effect exists [8, 9]. Different groups of users having, available, the same code and the same information for modeling an NPP do not achieve the same results. User effect is originated by the following:

(i) nodalization development, see also item (N), below;

(ii) interpreting the supplied (or the available) information, usually incomplete; see also item (M) below and Figure 2 where the same (imperfect) information from experimentalists (pressure drops across the steam generator are equal to -2.7 ± 5 KPa in the natural circulation test A2-77 performed in the LOBI facility) are correctly interpreted by the code users in different ways, thus generating (without surprise) different steady state results (see Figure 3);

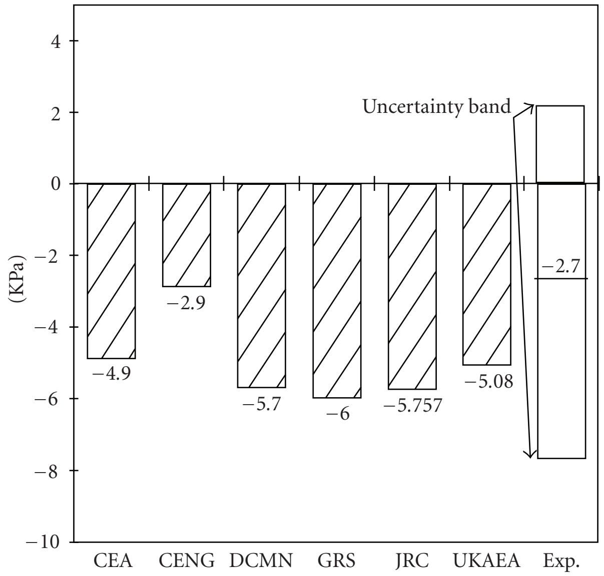

Figure 2: User effect in interpreting the available information (pressure drops across steam generator) from experiment (natural circulation test LOBI A2-77).

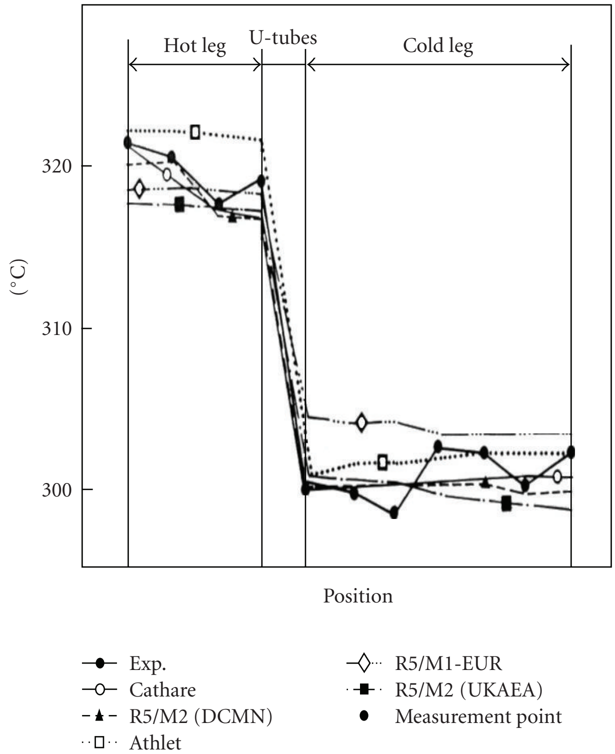

Figure 3: User effect in accepting (and qualifying) a steady state calculation (natural circulation test LOBI A2-77).

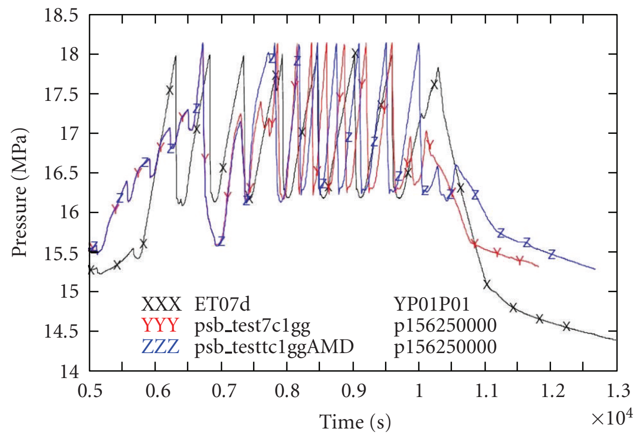

Figure 4: Computer/compiler effect: same code version and same input deck run on different computational platforms produces different results (reference to a qualified input-deck).

(iii) accepting the steady-state performance of the nodalization: code users must accept steady-state results before performing the transient analysis;. the "acceptance" of the steady-state results (Figure 3) "reflects" the choices made and affects (without surprise) the transient results;

(iv) interpreting transient results, planning and performing sensitivity studies, modifying the nodalization, and finally achieving "a reference" or "an acceptable" solution.

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

(L) The computer/compiler effect exists. A computer code is developed making use of the hardware selected by the code developers and available at the time when the code development starts. A code development process may last for a dozen of years, during which period, profound code hardware changes occur. Furthermore, the code is used on different computational platforms, and the current experience is that the same code with the same input deck applied within two computational platforms produces different results. Differences are typically small in "smoothly running transients," but may become noticeable in the case of threshold- or bifurcation-driven transients. Figure 4 depicts the comparison between the primary side pressure, during the PORV cycling period, of two calculations performed using exactly the same input deck and running on different computer configurations: the calculation labeled "psb_test7c1gg" has been run using a P-IV, 32 bits, 2800 MHz processor and Windows 32 bits as the operating system; the calculation labeled as "psb_testtc1ggAMD" has been run adopting an AMD Athlon, 64 bits 3200 + 2200 MHz as aprocessor and Windows 32 bites as the operating system. The experimental results are also added.

(M) The nodalization effect exists. The nodalization is the result of a wide range brainstorming process, where user expertise, computer power, and code manual play a role. There is a number of required code input values that cannot be covered by logical recommendations: the user expertise needed to fix those input values (e.g., Figure 2) may reveal inadequate and constitutes the origin of a specific source of

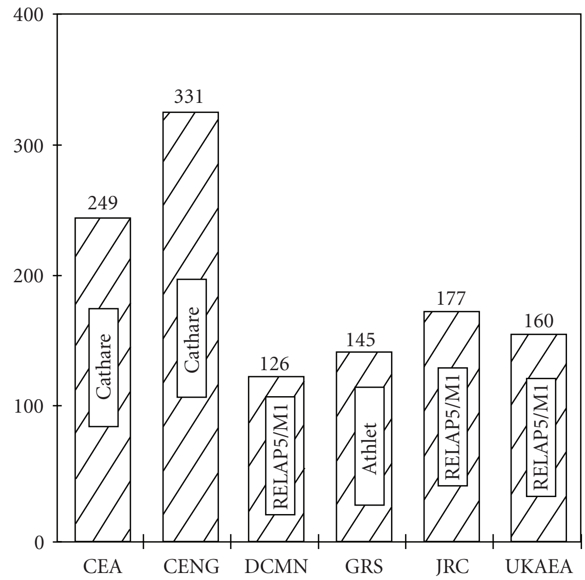

Figure 5: Nodalization effect: same facility (LOBI, natural circulation test A2-77) modeled with different number of control volumes.

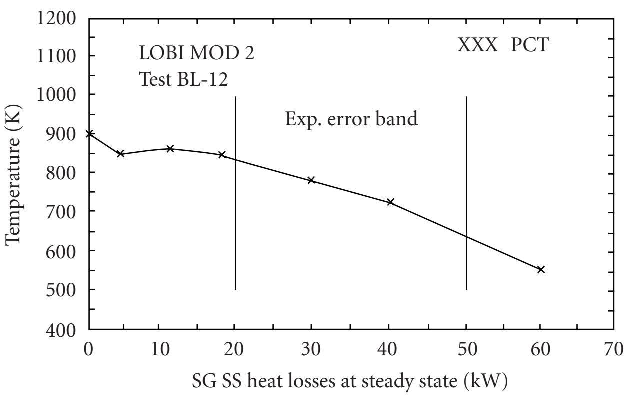

Figure 6: Imperfect knowledge of boundary conditions: effect of steam generator secondary side heat losses on the prediction of the peak cladding temperature (SBLOCA BL-12 in tLOBI-MOD 2 facility).

uncertainty. Figure 5 shows how the same facility (LOBI) is modeled with a different level of detail (i.e., number of control volumes) by the code users using either the same or different codes.

(N) *Imperfect knowledge of boundary and initial conditions.* Some boundary and initial conditions values are unknown or known with approximation: the code user must add information. This process unavoidably causes an impact on the results, which is not easily traceable and constitutes a specific source of uncertainty. Figure 6 constitutes an evident example of how the imperfect knowledge of the steam-generator secondary-side-heat losses (between 20 kW and 50 kW for the SBLOCA BL-12 performed in the LOBI-MOD 2 facility) has a strong impact (about 200 K) on the prediction of the peak cladding temperature (PCT).

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

(i) the heat transfer between the free liquid surface and the upper-gas-steam space;
(ii) the heat transfer between a hotter wall and the cold liquid down-flowing inside a steam-gas-filled region.

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

2.1. Code uncertainty

A system thermal-hydraulic code is a computational tool that typically includes three different sets of balance equations (of energy, mass, and momentum), closure or constitutive equations, material, and state properties, special process or component models, and a numerical solution method. Balance equations are not sophisticated enough for application in special components or for the simulation of special processes. Examples for those components are the pumps and the steam separators, and examples for those special processes are the countercurrent flow-limiting (CCFL) condition and the two-phase critical flow, though this is not true for all the codes. Empirical models "substitute" the balance equations in such cases. The sources of uncertainty connected with the code are those identified as (A) to (I) and (O) in the above list. Namely, the following association between uncertainty sources and code parts applies:

(i) balance equations, uncertainty sources (A) to (F);
(ii) closure and constitutive equations, uncertainty sources (H) and (I);
(iii) material properties, uncertainty source (J);
(iv) special process and component models, uncertainty sources (H), (I), and (O);
(v) numeric, uncertainty source (G).

2.2. Representation uncertainty

The representation uncertainty deals with the process of setting up the nodalization (idealization). The nodalization constitutes the connection between the code and the "physical reality" that is the objective of the simulation. The process for setting up the nodalization is a brainstorming activity carried out by the group of code users that aims at transferring the information from the real system (e.g., the NPP), including the related boundary and initial conditions, into a form understandable to the code. Limitation in available resources (in terms of man-months), lack of data, target of the code application, capabilities/power of the available computational machines, and expertise of the users have a role in this process. The result of the process may heavily affect the response of the code.

The source of uncertainty connected with the nodalization is identified as (M) in the above list, but the (J) source can also have a role.

2.3. Scaling issue

Scaling is a broad term used in nuclear-reactor technology as well as in basic fluid dynamics and in thermal hydraulics. 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 as well as adopted materials, including working fluids, and different ranges of variation for thermal-hydraulic quantities.

Therefore, the word "scaling" may have different meanings in different contexts. In system thermal hydraulics, a scaling process, based upon suitable physical principles, aim at establishing a correlation between (a) phenomena expected in an NPP transient scenario and phenomena measured in smaller scale facilities or (b) 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 basic uncertainties. Making reference to the identified list, the sources of uncertainty connected with the scaling are those applicable to the balance equations, for example, identified as (A) to (I). More precisely uncertainty sources associated to the scaling are (A) to (E), (H), and (I).

2.4. Plant uncertainty

Uncertainty or limited knowledge of boundary and initial conditions and related values for an assigned NPP are reported as plant uncertainty. Typical examples are the pressurizer level at the start of the assigned transient, the thickness of the gap of the fuel rod, the conductivity of the UO2, as well as the gap itself.

It might be noted that quantities like gap conductivity and thickness are relevant for the prediction of safety parameters (e.g., PCT) and are affected by other parameters like burn up whose knowledge is not as much detailed (e.g., each layer of a fuel element that may be part of the nodalization) as required. Thus such a source of error 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 as (N) in the above list.

2.5. User effect

Complex systems codes such as RELAP5, CATHARE, TRAC, and ATHLET have many degrees of freedom that allow misapplication (e.g., not using the countercurrent flow-limiting model at a junction where it is required) and errors by users (e.g., inputting the 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, consequently, will likely take different paths to obtain a problem solution. The cumulative effect of user community members to produce a range of answers using the same code for a well-defined problem with rigorously specified boundary and initial conditions is the user effect. The sources of uncertainty connected with the code user are those identified as (K) and (J). The code user has part of the responsibility associated with the source of uncertainty (L).

3. APPROACHES FOR COMPUTING UNCERTAINTY

In this section, the salient features of independent approaches for estimating uncertainties associated with predictions of complex system codes are reviewed as follows.

(i) *The propagation of code input errors (Figure 7).* This can be evaluated as being the most adopted procedure nowadays, endorsed by industry and regulators. It adopts the statistical combination of values from selected input uncertainty parameters (even though, in principle, an unlimited number of input parameters can be used) to calculate the propagation of the errors throughout the code.

(ii) *The propagation of code output errors (Figure 8).* This is the only demonstrated independent working alternative to the previous one and has also been used for industrial applications. It makes full and direct reference to the experimental data and to the results from the assessment process to derive uncertainty. In this case, the uncertainty prediction is not propagated throughout the code.

(iii) A third and independent way, that is, different from propagation of code input errors or from propagation of code output errors, (Figure 9) is based on adjoint sensitivity-analysis procedure (ASAP), global adjoint sensitivity-analysis procedure (GASAP) [10, 11], and data adjustment and assimilation (DAA) methodology [12] by which experimental and calculated data, including the computation of sensitivities (derived from ASAP), are mathematically combined for the prediction of the uncertainty scenarios.

The first approach, reviewed as the prototype for propagation of code input errors, is the so-called "GRS method" [13], which includes the so-called "CSAU method" (code scaling, applicability, and uncertainty) [4] and the majority of methods adopted by the nuclear industry. Although the entire set of the actual number of input parameters for a typical NPP input deck, ranging up to about 10^5 input parameters, could theoretically be considered as uncertainty sources by these methods, only a "manageable" number (of the order of several tens) is actually taken into account in practice. Ranges of variations, together with a suitable PDF (probability density function), are then assigned for each of the uncertain input parameter actually considered in the analysis. The number of computations, needed for obtaining the desired confidence in the results, can be determined

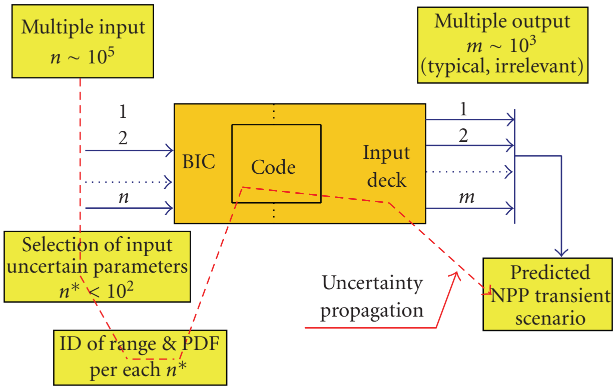

FIGURE 7: Uncertainty methods based upon propagation of input uncertainties.

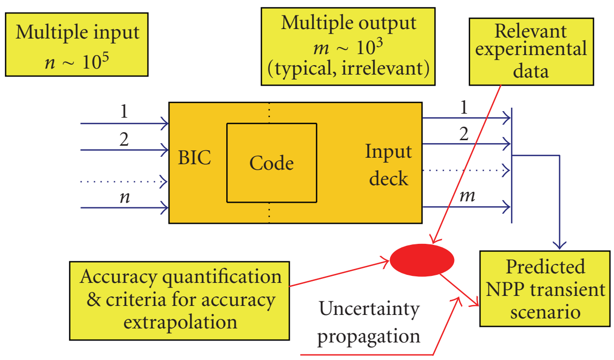

FIGURE 8: Uncertainty methods based upon propagation of output uncertainties.

theoretically by the Wilks formula [14]. Subsequently, the identified computations (ca. 100) are performed using the code under investigation to propagate the uncertainties inside the code, from inputs to outputs (results). The logical steps of the approach are depicted in Figure 7. The main drawbacks of such methods are connected with (a) the need of engineering judgment for limiting (in any case) the number of the input uncertain parameters, (b) the need of engineering judgment for fixing the range of variation and the PDF for each input uncertain parameter, (c) the use of the code-nodalization for propagating the uncertainties, if the code-nodalization is wrong, not only the reference results are wrong but also the results of the uncertainty calculations (see sources of uncertainty (L), (M), and (O) in Section 2 and the first TRUE in Section 4), and (d) the process of selecting the (about) 100 code runs is demonstrably not convergent, and the investigation of results from two or more different sets of 100 calculations shows different values for uncertainty. A support to the last consideration is supplied by Figure 10 that summarizes a study performed by Kaeri in the framework of the Phase III of BEMUSE project [7]. A direct Monte Carlo simulation consisting of 3500 runs was performed for simulating the LBLOCA L2-5 in the LOFT facility, and several samples of $n = 59$ and $n = 93$ calculations were considered. The following considerations apply.

(i) From about 1000 runs, the mean value (equal to 1034 K) and the 95% empirical quantile (equal to 1173 K) of the first PCT are almost stabilized.

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

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

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

  (b) At first order, among the 58 calculations ranging from 1170 K to 1360 K, none was found significantly lower than the 95% quantile of the 3500 code runs, notwithstanding, statistically three cases (i.e., 5% of 58) are expected.

  (c) At the second order, among 37 calculations ranging from 1150 K to 1270 K, one case was found below 1173 K.

The second approach (Figure 8), reviewed as the propagation of code output errors, is representatively illustrated by the UMAE-CIAU (uncertainty method based upon accuracy extrapolation [15] "embedded" into the code with capability of internal assessment of uncertainty [16, 17]). Note that this class of methods includes only a few applications from industry. The use of this method depends on the availability of "relevant" experimental data, where, here, the word "relevant" is connected with the specific NPP transient scenario under investigation for uncertainty evaluation. Assuming such availability of relevant data, which are typically ITF (integral test facility) data, and assuming that the code correctly simulates the experiments, it follows that the differences between code computations and the selected experimental data are due to errors. If these errors comply with a number of acceptability conditions [15], then the resulting (error) database is processed and the "extrapolation" of the error takes place. Relevant conditions for the extrapolation are (i) building up the NPP nodalization with the same criteria as was adopted for the ITF nodalizations, and (ii) performing a similarity analysis and demonstrating that NPP calculated data are "consistent" with the data measured in a qualified ITF experiment.

The main drawbacks of this method are as follows.

(i) The method is not applicable in the absence of relevant experimental information.

(ii) A considerable amount of resources is needed to establish a suitable error database, but this is a one-time effort, independent of subsequent applications of this method.

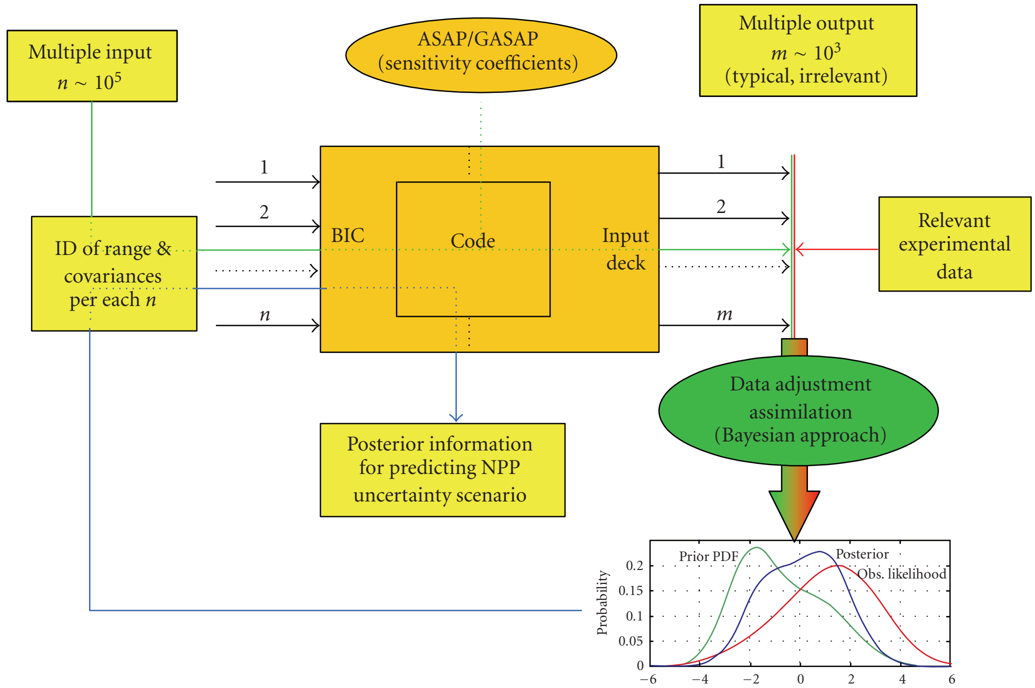

FIGURE 9: Uncertainty methodology based on adjoint sensitivity analysis procedure and data adjustment/assimilation.

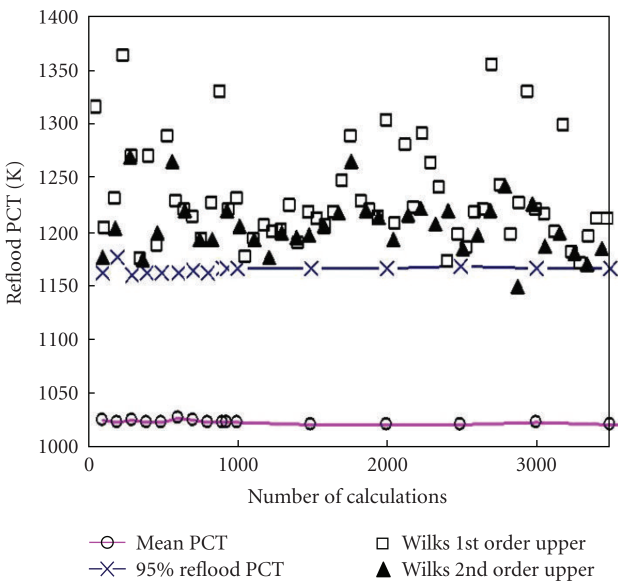

FIGURE 10: KAERI direct Monte-Carlo analysis performed in the framework of Phase III of BEMUSE project: spread of the upper limit of PCT using Wilks' formula at first and second order.

(iii) The process of combining errors originating from different sources (e.g., stemming from different ITF or SETF (Separate Effect Test Facility), different but consistent nodalizations, and different types of transient scenarios) is not based upon fundamental principles and requires detailed validation.

The third approach, depicted in Figure 9, is based upon the powerful mathematical tools of ASAP, GASAP and, DAA, by which all parameters $\boldsymbol{\alpha}$ that affect any prediction, being part of either the code models or the input deck can be considered. The adjoint sensitivity-analysis procedure (ASAP) [10, 11] is the most efficient deterministic method for computing local sensitivities $\mathbf{S}$ of large-scale systems, when the number of parameters and/or parameter variations exceeds the number of responses $\mathbf{R}$ of interest (that is the case of most problems of practical interest). In addition, also the system critical points $\mathbf{y}$ (i.e., bifurcations, turning points, saddle points, and response extrema) can be considered and determined by the global adjoint sensitivity-analysis procedure (GASAP) [10, 11] in the combined phase space formed by the parameters, forward state variables, and adjoint variables. Subsequently, the local sensitivities of the responses $\mathbf{R}$ located at critical points $\mathbf{y}$ are analyzed by the efficient ASAP.

Once the sensitivity matrix $\mathbf{S}$ of the responses $\mathbf{R}$, with respect to the parameters $\boldsymbol{\alpha}$, is available, the moment propagation equation is adopted to obtain the computed covariance matrix $\mathbf{C_R}$ of the responses starting from the covariance matrix $\mathbf{C}_{\boldsymbol{\alpha}}$ of the system parameters. The elements of the matrix $\mathbf{C}_{\boldsymbol{\alpha}}$ reflect the state of knowledge about the input (uncertainty) parameters that can be characterized by ranges and PDFs. It is very well known that, in system thermal hydraulics only few elements of $\mathbf{C}_{\boldsymbol{\alpha}}$ are obtained from experimental observations (mainly from SETF), whereas, for the major part of them, engineering judgment is adopted for deriving "first"-guess values of ranges and PDFs. The imperfect knowledge of the input uncertainty parameter obviously affects the computed responses $\mathbf{R}$ and the relative covariance $\mathbf{C_R}$ and constitutes the main reason, for which, proper experimental data (i.e., connected with the specific NPP transient scenario under investigation for uncertainty evaluation) are needed. The technique, by which, experimental observations are combined with code predictions and their respective errors to provide an improved estimate of the system state is known as data adjustment and assimilation (DAA) [12], and it is based on a Bayesian inference process.

The idea at the basis of DAA can be made more specific as follows. The computed results $\mathbf{R}$ and the respective statistical errors $\mathbf{C_R}$ predicted by mathematical models and based on prior- or first-guess PDFs for the input parameters (i.e., $\mathbf{C}_{\boldsymbol{\alpha}}$) are combined with proper experimental observations $\mathbf{M}$ of the states of a system to generate "adjusted" values for the system parameters ($\boldsymbol{\alpha}^{\mathrm{IE}}$, where the suffix IE stays for improved estimate values) and the respective input covariance matrix ($\mathbf{C}_{\boldsymbol{\alpha}}^{\mathrm{IE}}$, or "posterior" PDFs). From this process, which can be considered as improved estimate analysis of the system states, the responses $\mathbf{R}^{\mathrm{IE}}$ and the respective covariance matrix ($\mathbf{C}_{\mathbf{R}}^{\mathrm{IE}}$) are finally derived.

In conclusion, to reduce uncertainties in both the system parameters and responses, the Bayesian inference procedure is used to consistently assimilate computational and experimental information. There are several approaches possible when performing a DAA process in conjunction with time-dependent nonlinear systems, but the "online data adjustment/assimilation" is the best suited for uncertainty analysis of large-scale highly nonlinear time-dependent problems. It can be performed online (i.e., sequentially in time and interactively with the code that calculates the system dependent variables and responses) by decomposing the original system into simpler but interacting subsystems. In the present case, the assimilation process involves, at every time node, the minimization of a quadratic objective function subject to constraints.

Once a suitable database of improved estimates for the input parameters ($\boldsymbol{\alpha}^{\mathrm{IE}}$) and for the respective input covariance matrix ($\mathbf{C}_{\boldsymbol{\alpha}}^{\mathrm{IE}}$) is available, the application of the method to an NPP scenario is straightforward and requires (a) the calculation of the reference responses $\mathbf{R}^{\mathrm{NPP}}$, where, here, the word "reference" is connected with the reference NPP boundary and initial conditions supplemented by improved estimates of the input parameters ($\boldsymbol{\alpha}^{\mathrm{IE}}$) when other information are not available, (b) the computation of the sensitivity coefficients $\mathbf{S}$, and (c) the application of the moment-propagation equation to obtain the computed covariance matrix $\mathbf{C}_{\mathbf{R}}^{\mathrm{NPP}}$ of the responses starting from the covariance matrix $\mathbf{C}_{\boldsymbol{\alpha}}^{\mathrm{NPP}}$ of the system parameters supplemented by improved estimates of the input covariance matrix ($\mathbf{C}_{\boldsymbol{\alpha}}^{\mathrm{IE}}$) when other information are not available.

The main drawbacks of this approach are as follows.

(i) The method is not applicable in the absence of relevant experimental information.

(ii) The adjoint model, needed for computing the sensitivity $\mathbf{S}$, requires relatively modest additional resources to develop and implement if this is done simultaneously with the development of the original code; however, if the adjoint model is constructed a posteriori, considerable skills may be required for its successful development and implementation.

(iii) A considerable amount of resources is needed to establish a suitable database of improved estimates for the input parameters ($\boldsymbol{\alpha}^{\mathrm{IE}}$) and for the respective input covariance matrix ($\mathbf{C}_{\boldsymbol{\alpha}}^{\mathrm{IE}}$), but this is a one-time effort, independent of subsequent applications of the method.

The maturity of the methods at the first two items may be considered as proved and also based upon applications completed within the framework of initiatives of international institutions (OECD/NEA [5–7] and IAEA). The reason for the consideration of the approach at the third item derives from its potential to open an independent way (i.e., different from propagation of code input errors or from propagation of code output errors) for performing global uncertainty analysis. In this case, the method itself, as an uncertainty procedure, is not an established technology, but it constitutes an established idea and framework to pursue a mathematically based road to evaluate the uncertainty in system-code predictions.

In the following subsections, short descriptions of the GRS and UMAE methods are given for the sake of completeness. More detailed information about the CIAU methodology, based on (code) internal assessment of uncertainty and on propagation of output errors, are given in Section 5.

3.1. GRS method

The GRS method [13] is a probabilistic method based on the concept of propagating the input uncertainties. All relevant uncertain parameters including the code, representation, and plant uncertainties are identified, any dependencies between uncertain parameters are quantified and ranges and/or PDFs for each uncertain parameter are determined. Expert judgment and experience from code applications to separate and integral test and full plant application are principal sources of information for uncertain parameters identification and quantification. Peculiarities of the GRS method are as follows.

(i) The uncertainty space of input parameters (defined by their uncertainty ranges) is sampled at random according to the combined "subjective" probability distribution of the uncertain parameters, and code calculations are performed by sampled sets of parameters.

(ii) The number of code calculations is determined by the requirement to estimate a tolerance-confidence interval for the quantity of interest (such as peak-clad temperature). The Wilks formula [14] is used to determine the number of calculations needed for deriving the uncertainty bands.

(iii) Statistical evaluations are performed to determine the sensitivities of input parameter uncertainties on the uncertainties of key results (parameter-importance analysis).

(iv) There are no limits for the number of uncertain parameters to be considered in the analysis, and the calculated uncertainty has a well-established statistical basis.

(v) The method relies only on actual code calculations without using approximations like fitted response surfaces.

For the selected plant transient, the method is applied to an integral-effects test 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 modeling uncertainties and the related parameters, and identify and quantify dependencies between uncertain parameters. Subjective probability density functions are used to quantify the state of knowledge of uncertain parameters for the specific scenario. The term "subjective" is used here to distinguish uncertainty due to imprecise knowledge from uncertainty due to stochastic or random variability.

Uncertainties of code model parameters are derived based on validation experience. The scaling effect has to be quantified as model uncertainty. Additional uncertain model parameters can be included, or PDF can be modified, accounting for results from the analysis of separate-effects tests.

Input parameter values are simultaneously varied by random sampling according to the subjective PDF and dependencies. A set of parameters is provided to perform the required number n of code runs. For example, the 95% fractile and 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 or needed.

Sensitivity measures by using regression or correlation techniques from the sets of input parameters and from the corresponding output values allow the ranking of the uncertain input parameters in relation to their contribution to output uncertainty. Therefore, the ranking of parameters is a result of the analysis, not of prior expert judgment. 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 PCT is below the tolerance limit with at least $\alpha = 95\%$ probability). One can be $\beta\%$ confident that at least $\alpha\%$ 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 the Wilks formula

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

and is representing 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. For two-sided statistical tolerance intervals (investigating the output parameter distribution within an interval), the formula is

$$1-\alpha^n - n\cdot(1-\alpha)\cdot\alpha^{n-1} \geq \beta. \tag{2}$$

The minimum number n of calculations for both one sided and two sided can be found in Table 1. 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 deriving sensitivity measures is also independent of the number of parameters. As an example, a total number of 100 runs is typical for the application of the GRS method. 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 should be performed.

3.2. UMAE method

The UMAE [15], whose flow diagram is given in Figure 11, is the prototype method for the description of "the propagation of code-output-errors" approach. The method focuses not on the evaluation of individual parameter uncertainties but on the propagation of errors from a suitable database calculating the final uncertainty by extrapolating the accuracy from relevant integral experiments to full scale NPP.

Considering ITF of reference water-cooled reactor and qualified computer codes based on advanced models, the method relies on code capability, qualified by application to facilities of increasing scale. Direct data extrapolation from small-scale experiments to reactor scale is difficult due to the imperfect scaling criteria adopted in the design of each scaled down facility. So only the accuracy (i.e., the difference between measured and calculated quantities) is extrapolated. Experimental and calculated data in differently scaled facilities are used to demonstrate that physical phenomena and code predictive capabilities of important phenomena do not change while increasing the dimensions of the facilities (see right loop FG in Figure 11).

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

The method utilizes a database from similar tests and counterpart tests performed in ITF, that are representative of plant conditions. The quantification of code accuracy (step f in Figure 11) is carried out by using a procedure based on the fast Fourier transform-based method (FFTBM) [18], characterizing the discrepancies between code calculations and experimental data in the frequency domain, and defining figures of merit for the accuracy of each calculation. Different requirements have to be fulfilled in order to extrapolate the accuracy.

Calculations of both ITF experiments and NPP transients are used to attain uncertainty from accuracy. Nodalizations

Table 1: GRS method: number of minimum calculations.

	One-sided statistical tolerance limit			One-sided statistical tolerance limit		
β/α	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

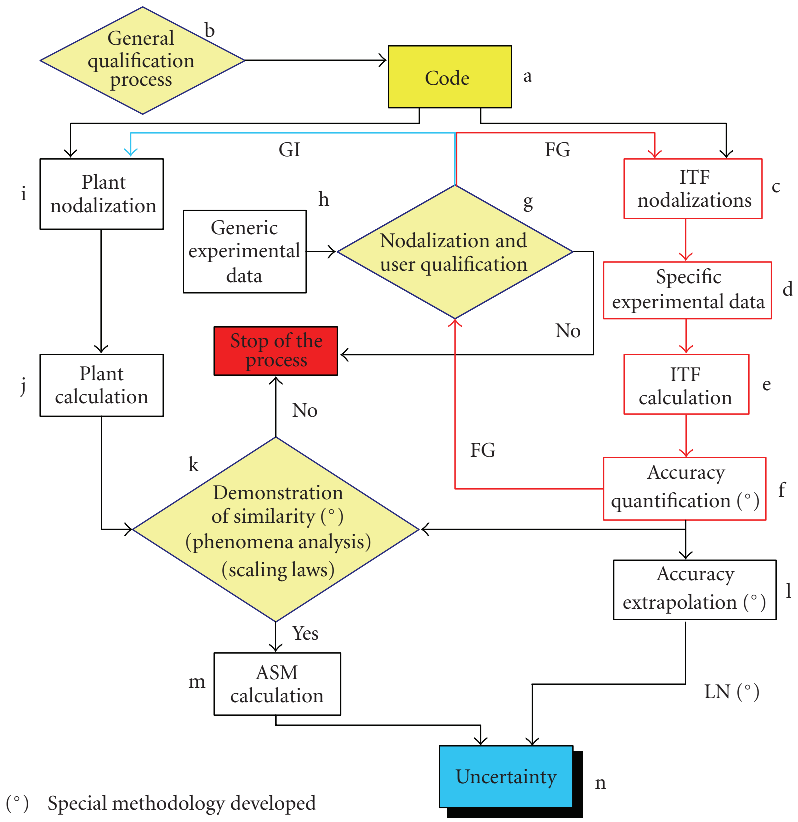

Figure 11: UMAE flow diagram (also adopted within the process of application of CIAU).

are set up and qualified against experimental data by an iterative procedure, requiring that a reasonable level of accuracy is satisfied. Similar criteria are adopted in developing plant nodalization and in performing plant transient calculations (see left loop FG in Figure 11). The demonstration of the similarity of the phenomena exhibited in test facilities and in plant calculations, accounting for scaling-laws considerations (step "k" in Figure 11), leads to the analytical simulation model (ASM), that is, a qualified nodalization of the NPP.

4. RELEVANT TOPICS FOR UNCERTAINTY EVALUATION

Fundamental aspects to be considered when developing an uncertainty method are briefly presented [19]. The definition of "topics relevant for uncertainty evaluation," with the acronym TRUE, is introduced to emphasize the central role they have to play in structuring the architecture of a methodology. The following three TRUEs are discussed, and for each of them, one example is given, together with the lesson learned.

(i) *The nodalization choices.* Different input decks (i.e., nodalization user choices) produce different effects upon relevant code output parameters.

(ii) *The code versions.* Different code versions (same developer) have a strong impact on the prediction of relevant code-output parameters.

(iii) *The bifurcation analysis.* Scenarios can be imagined where bifurcations bring the transient evolution far from the best-estimate deterministic prediction, thus invalidating the connected uncertainty evaluation.

4.1. The nodalization choices

Results from the analysis of the LBLOCA DEGB in the Angra 2 NPP are considered [20]. A "fictitious" 3D nodalization of the reactor pressure vessel (RPV) was adopted, and the influence of upper-plenum (UP) noding assumption was considered by developing three different RPV-UP

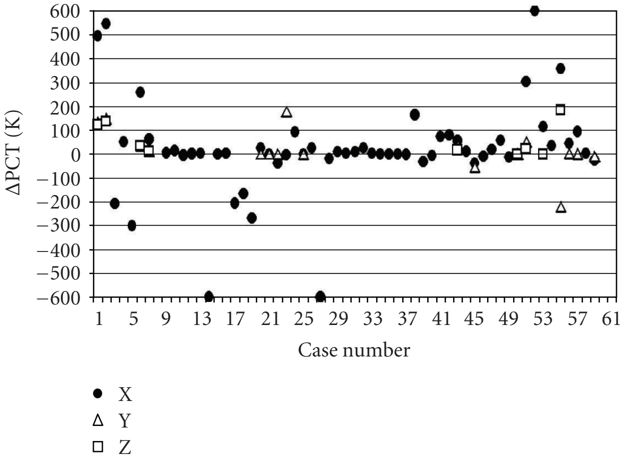

Figure 12: TRUE: influence of the nodalization choices.

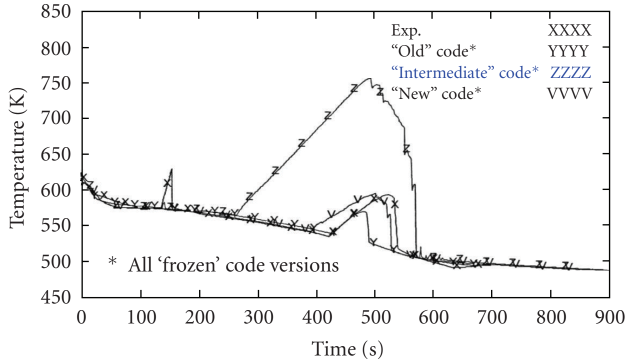

Figure 13: TRUE: influence of the code versions.

nodalizations simulating one-uniform (Z in Figure 12) and two-nonuniform (X and Y in Figure 12) UP behaviors (top-down flow allowed in all channels, top-down flow allowed in all channels except in the hot assembly, and top-down flow allowed only in the determined breakthrough channels, resp.).

A comprehensive sensitivity study has been carried out, aiming at confirming the influence of selected input parameters upon the LBLOCA predicted scenario, and at showing the importance of nodalization upon the same prediction when an assigned input parameter is varied. Starting from the "reference" nodalizations (X, Y, and Z), single parameters are varied in each code run. Sixty one variations of input parameters, subdivided in six groups ("fuel," "nodalization," "loop hydraulics," "PSA and ECCS," "neutronics," and "other"), were considered.

The dispersion of results for ΔPCT (defined as the difference between the PCT of the reference calculation and the PCT obtained from the generic sensitivity run) can be seen in Figure 12. The following two outcomes can be detected:

(i) the reference PCT are affected by the nodalization (i.e., choices);

(ii) the ΔPCT are strongly affected by the nodalization (i.e., a given input uncertain parameter is relevant or not depending upon the selected nodalization). Moreover, also the sign of ΔPCT (i.e., the increase or decrease of the PCT value with respect to the reference calculation) is nodalization dependent (e.g., sensitivity case no. 55).

It should be noted that the conclusions at items (i) and (ii) are also applicable when different thermal-hydraulic system codes are adopted. The lesson learned, that is, the importance of the nodalization and of the code upon the predicted scenario, should be duly considered when the evaluation of the uncertainty of relevant code output parameters is performed by the process of propagating-input uncertainties through the code (i.e., propagation of code input uncertainties) that is affected by the code itself and by the nodalization.

4.2. *The code versions*

After the closure of the uncertainty-method study (UMS), University of Pisa performed comparison calculations of experiment LSTF-SB-CL-18 using different versions of the RELAP5 code, that is, in chronological order MOD 2, MOD 3.2, and MOD 3.2.2. In the UMS study, Mod 2 was used by the University of Pisa, and MOD 3.2 by AEA Technology as well as ENUSA. In the performed a posteriori analysis, it turned out that MOD 3.2 calculated a 170 K higher-peak-clad temperature compared with MOD 2 and MOD 3.2.2 using the same input deck (Figure 13). This is in agreement with the AEAT reference peak-clad temperature value and may contribute to the relative high upper limit of the uncertainty ranges calculated by AEAT and ENUSA in the framework of UMS.

The lesson learned from this TRUE is that the code versions (highly evaluated and qualified system thermal-hydraulic code), with the same input deck, have strong impact upon results, and affect uncertainty prediction. Therefore, "direct" specific code qualification is needed for uncertainty evaluation and the "internal assessment of uncertainty" (see Section 5), by which the uncertainty methodology is strictly connected with the code version, is a highly recommended property to consider.

4.3. *The bifurcation analysis*

Scenarios can be imagined where bifurcations bring the transient evolution far from the best-estimate deterministic prediction, thus invalidating the connected uncertainty evaluation. Therefore, a bifurcation analysis may reveal necessary. Bifurcations can be originated by the actuation or lack of actuation of a system (e.g., pressurizer relief valves) or by the occurrence of a physical phenomenon characterized by a threshold (typically, the dryout). A tree of uncertainty bands can be predicted by CIAU, and the results of a sample application [21] can be seen in Figure 14. The CIAU-bifurcation capability was applied by University of Pisa in the post-UMS study and the uncertainty ranges obtained by AEAT (extreme results in the UMS framework) were (basically) reproduced by the CIAU bifurcation study. The lesson learned from this experience is that bifurcation study is possible and produces

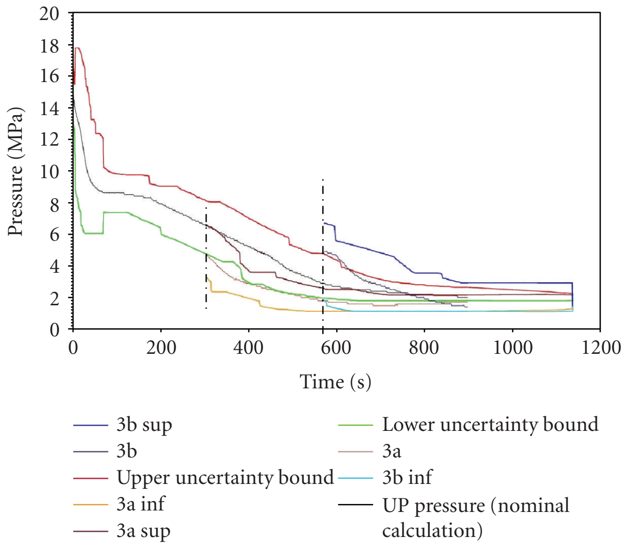

Figure 14: TRUE: consideration of bifurcation analysis performed by CIAU (tree of uncertainty bands).

(as expected) wider uncertainty bands compared with a standard uncertainty study.

5. THE INTERNAL ASSESSMENT OF UNCERTAINTY

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

(i) the resources needed for their application may be very demanding, ranging up to several man-years;
(ii) the achieved results may be strongly method-user dependent.

The last item should be considered together with the code-user effect, widely studied in the past, and may threaten the usefulness or the practical applicability of the results achieved by an uncertainty method. Therefore, the internal assessment of uncertainty (IAU) was requested as the follow-up of an international conference [22]. The approach CIAU, code with capability of IAU, has been developed with the objective of reducing the limitations discussed above.

5.1. CIAU method

The basic idea of the CIAU [16, 17] can be summarized in two parts.

(i) *Consideration of plant status.* Each status is characterized by the value of six "driving" (i.e., relevant) quantities (whose combination is called "hypercube") and by the time instant when those values are reached during the transient.
(ii) Association of uncertainty (quantity and time) to each plant status.

The key feature of CIAU is the continuous full reference to the experimental database. Accuracies detected from the comparison between experimental and calculated data are extrapolated to obtain uncertainty in the system-code predictions. A solution to the issues constituted by the "scaling" and "the qualification" of the computational tools is embedded into the method [23, 24], through the UMAE methodology, that constitutes the engine for the development of CIAU and for the creation of the error database.

Assigned a point in the time domain, the accuracy in predicting the time of occurrence of any point is distinguished from the accuracy that characterizes the quantity value at that point. Thus the time domain and the phase space are distinguished: the time domain is needed to characterize the system evolution (or the NPP accident scenario), and the phase-space domain is used to identify the hypercubes. The safety relevance and the consistency with the technological achievements have been considered in order to select the driving quantities, listed as (1) through (6) in Table 2, whose upper and lower boundaries have been fixed together with a minimum-optimal number of intervals determined considering the following aspects: (i) design of primary system plant, (ii) design and licensing of ECCS, (iii) design and optimization of emergency operational procedures, (iv) benchmarking of simplified models, (v) training purpose, and (vi) code limitations. About the transient time, a stable steady-state (or stationary) situation must occur, or be specified, when a code calculation is concerned.

Quantity and time accuracies can be associated to errors-in-code models and uncertainties in boundary and initial conditions including the time sequence of events and the geometric modeling (or nodalization) of the problem. Consider the following facts.

(i) The "transient-time-dependent" calculation by a code resembles a succession of steady-state values at each time step and is supported by the consideration that the code uses, and is based on a number and a variety of empirical correlations valid (and qualified) at a steady state with assigned geometric discretization (or nodalization) for the concerned system. Therefore, quantity accuracy can be associated primarily with errors-in-code models.
(ii) Error associated with the opening of a valve (e.g., time when the equivalent full-flow area for the flow passage is attained) or inadequate nodalization induce time errors that cannot be associated to code-model deficiencies. Therefore, time accuracy can be associated primarily with uncertainties-in-boundary and initial conditions.

Once the time-accuracy (uncertainty) vector (TAV, TUV) and the quantity-accuracy (uncertainty) matrix (QAM, QUM) are derived, the overall accuracy (and uncertainty) is obtained by the geometric combination of the two accuracies (and uncertainties) values, that is, time and quantity, in the two-dimensional space-time plane.

A general idea of the architecture of the CIAU methodology can be derived from Figure 15. Mainly two processes can be distinguished, the "error-filling process" by which the NPP statuses are filled with the values of the error database, and the "error-extraction process" by which the uncertainty values (derived from the extrapolation process of accuracy)

Table 2: CIAU method: subdivision of driving quantities into intervals.

PWR-DRIVING QUANTITIES		(1) Upper plenum pressure (MPa)	(2) Primary circuit mass inventory (%)[(a)]	(3) Steam generator pressure (MPa)	(4) Cladding temperature (K)	(5) Core power (%)[(a)]	(6) Steam generator level (%)[(a)]
Hypercube intervals	1	0.09–0.5	10–40	0.1–3.0	298–473	0.5–1.0	0–50
	2	0.5–2.0	40–80	3.0–7.0	473–573	1.0–6.0	50–100
	3	2.0–4.0	80–100	7.0–9.0	573–643	6.0–50	100–150
	4	4.0–5.0	100–120	—	643–973	50–100	—
	5	5.0–7.0	—	—	973–1473	100–130	—
	6	7.0–9.0	—	—	—	—	—
	7	9.0–10.0	—	—	—	—	—
	8	10.0–15.0	—	—	—	—	—
	9	15.0–18.0	—	—	—	—	—

(a) Percent of the initial (nominal) value.

are picked up from the NPP statuses selected during the transient calculation to generate continuous uncertainty bands enveloping the ASM, that is, the qualified NPP calculation in the UMAE nomenclature.

Summarizing, six dimensions constitute the phase-space domain and each combination of intervals of the driving quantities identifies one hypercube in that domain. Therefore, a hypercube and a time interval characterize a unique plant status in the frame of uncertainty evaluation and all plant statuses are characterized by a matrix of hypercubes and by a vector of time intervals. Each point of the curve (generic thermal-hydraulic code output plotted versus time) 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 the time uncertainty. 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 to be acceptable to the USNRC staff for comparison of best-estimate (BE) predictions of postulated transients to the licensing limits in 10 CFR Part 50.

One main difference between UMAE an CIAU has to be emphasized: in the UMAE methodology, the uncertainty of thermal-hydraulic quantity is an average of the values obtained in different simulations of the same class of transients and in the same facility or in similar tests performed in different facilities; in the case of CIAU, the results of any kind of transients can be combined to derive the accuracy and then the uncertainty if they pass through the same plant status (i.e., hypercube and time when the hypercube is reached).

5.2. Qualification processes in CIAU method

One important aspect of any tool developed in system thermal hydraulics is the possibility to perform an assessment and eventually to show the quality level, utilizing databases independent of those utilized in the development of the tool itself. Two qualification steps are foreseen in the case of CIAU.

The first one can be identified as the internal qualification process. Data gathered inside each hypercube or each time interval of QUM and TUV for uncertainty evaluation, or inside QAM and TAV for accuracy evaluation, are labeled before being combined. In other terms, each uncertainty- or accuracy-connected value includes its origin, that is, the transient scenario type and the part of the hypercube that is concerned. A statistical analysis can be used to find whether groups of data, coming from different events or related to different parts of the same hypercube, are different. For instance, it might happen that data from the analysis of several SBLOCAs produce uncertainty values much higher than data from the analysis of a similar number of LBLOCAs, when the same hypercubes are concerned. In this case, the number of hypercubes, that is, the ranges of variation of the driving quantities, must be changed or the transient type must be identified inside each hypercube. More in detail, it must be shown that accuracy and uncertainty values in each hypercube or in each time interval do not depend upon

(i) time (into the transient) when the hypercube is reached;
(ii) volume scaling factors;
(iii) transient type (e.g., SBLOCA, LBLOCA, LOFW, etc.);
(iv) dimension of hypercubes;
(v) ITF, SETF or NPP characteristics.

Example of how the internal qualification process [25] is lead is given in Figure 16. For reason of simplicity, here, the analysis is focused only on the accuracy values of one quantity inside one hypercube. The selected quantity is the cladding temperature, whereas the selected hypercube is one of those containing the higher number of experimental transients (hypercube 9-4-2-3-5-3, where each digit varies between 1 and the number of intervals by which each "driving" quantity is subdivided, see Table 2). Figure 16 shows that inside the selected hypercube, no correlations based on the transient type and the on-volume scaling factor (Kv) can be established among the accuracy values of the cladding temperature.

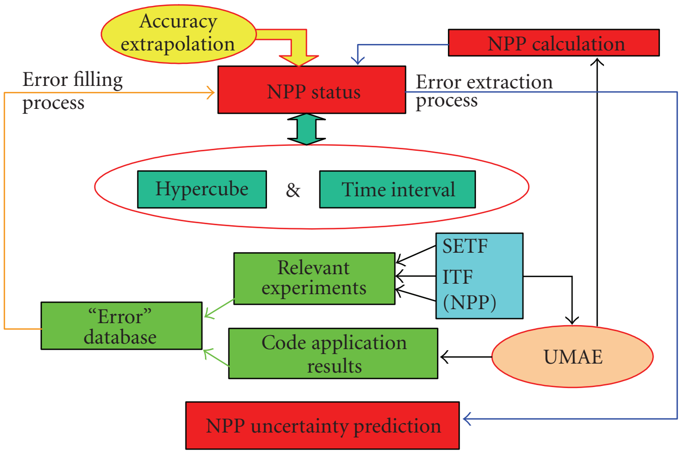

FIGURE 15: CIAU method: "error filling process" and "error extraction process."

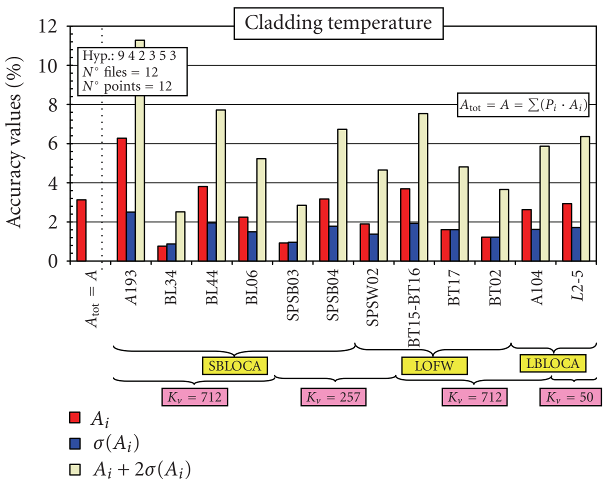

FIGURE 16: CIAU method: independence of accuracy on transient type & Kv ($A_{TOT} = A$ is the weighted—through the weights P_i—cladding temperature accuracy inside the hypercube 9-4-2-3-5-3 derived from the (12) average accuracy values (A_i) of the experimental transient i).

The internal qualification process is continuously ongoing during the development of the method: the experience gained, so far, does not bring to any 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 ITF that have not been utilized for getting uncertainty values. The success consists in demonstrating that CIAU calculated uncertainty bands envelop the experimental data. This must be intended as the reference (external) qualification process for the CIAU, together with the condition that uncertainty bands are reasonably large. The completion of this step will also allow establishing, on an objective basis, the confidence level of the uncertainty statements. 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, though the target is to achieve the 95% confidence level.

6. CONCLUSIONS

The uncertainty evaluation constitutes the ending necessary step for the application of a system thermal-hydraulic code to the nuclear technology. Therefore, any application of a best estimate code without the uncertainty evaluation is meaningless simply because an error is unavoidable for any prediction. The nuclear safety principles and, primarily, the concepts like defense in depth are the reasons why an uncertainty analysis is performed. It must be ensured that the nominal result of a code prediction, "best estimate" in the present case, is supplemented by the uncertainty statement, that can be simplified as "uncertainty bands," in such a way that connected safety margins are properly estimated.

Several sources of uncertainty have been classified (first goal of the paper) and topics relevant for uncertainty evaluation have been emphasized (second goal) to investigate which of these are embedded into the currently adopted methods and which comes out in the frame of their applications. The third purpose of the paper is twofold: (a) to identify the roadmaps for uncertainty evaluation adopted by the methods currently applied to the cases of industrial interest, making reference to the classification based on propagation of code input errors and propagation of code output errors, and (b) to propose an innovative method (based on the adjoint and global adjoint sensitivity-analysis procedure extended to performing uncertainty evaluation in conjunction with data adjustment and assimilation) that might not suffer from the drawbacks identified for the current methods.

Finally, a method to calculate the uncertainty associated with NPP computer-code calculations directly integrated in the code has been presented. The main advantage of an IAU approach consists in avoiding, from the methodology user point of view, the interpretation of logical statements that are part of the application process for all current uncertainty methods, that is, avoiding user effect when using uncertainty methodologies. The above consideration does not exclude the use of engineering judgment: rather, engineering judgment is embedded into the development of the IAU method and is not needed in its application.

LIST OF ABBREVIATIONS

ASAP:	Adjoint sensitivity-analysis procedure
ASM:	Analytical simulation model
BE:	Best estimate
BEMUSE:	Best-estimate methods uncertainty and sensitivity evaluation
CCFL:	Counter current-flow limiting
CIAU:	Code with the capability of internal assessment of uncertainty
CSAU:	Code scaling, applicability, and uncertainty evaluation
DAA:	Data adjustment and assimilation
DEGB:	Double-end guillotine break
ECCS:	Emergency core-cooling system
FFTBM:	Fast Fourier transform-based method
GASAP:	Global adjoint sensitivity-analysis procedure
IAU:	Internal assessment of uncertainty
ITF:	Integral test facility
LBLOCA:	Large break loss-of-coolant accident
LOCA:	Loss-of-coolant accident
LOFW:	Loss-of-feed water
NPP:	Nuclear power plants
PCT:	Peak cladding temperature
PDF:	Probability density function
PORV:	Pilot-operated relief valve
PSA:	Probabilistic safety analysis
QAM:	Quantity accuracy matrix
QUM:	Quantity uncertainty matrix
RPV:	Reactor pressure vessel
SBLOCA:	Small break loss-of-coolant accident
SETF:	Separate effect-test facility
TAV:	Time accuracy vector
TRUE:	Topics relevant for uncertainty evaluation
TUV:	Time uncertainty vector
UMAE:	Uncertainty methodology based on accuracy extrapolation
UMS:	Uncertainty methods study
UP:	Upper plenum.

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Review Article

BWR Stability Issues in Japan

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The present paper reviews activates relevant to the boiling water reactor (BWR) stability phenomenon, which has a coupled neutronic and thermal-hydraulic nature, from the viewpoint of model and code developments and their applications to the BWR stability solution methodology in Japan.

1. INTRODUCTION

The core power oscillation phenomenon inherently exists in BWR cores [1], as generally called by the BWR stability or instability. The BWR instability is possible even at the normal plant operation conditions, and significant core power oscillations may threaten core fuel integrity due to the fuel cladding dryout occurrence and/or due to the strong PCMI (pellet-cladding mechanical interaction). Therefore, an accurate prediction for the onset of BWR instability is indispensable for the safety of BWR core design and operation. Hence, numerous efforts have been paid to understand the complicated BWR instability mechanism and to develop the advanced analysis models.

The stability problem has become an important concern on safety of BWR operations, in particular, after the instability incident at LaSalle-2. It should be emphasized that the applied analysis code predicted a stable core condition while instability actually occurred. Therefore, GE and US BWROG (BWR Owners' Group) have improved the stability analysis models which can be adequately applicable to the actual core design and operation, and have developed the long-term stability solution methodologies with several modifications in the plant installation.

Also in Japan, similar activities have been proceeded by the BWR plant/fuel vendors and utilities to exclude any instability concern. Main goals in Japanese activities are as follows: (1) to analytically investigate the complicated BWR instability mechanism, the power oscillation onset/growth, and formation of the limit cycle oscillation, by using the three-dimensional time-domain code; (2) to empirically define the stability performance of the employed fuel design, and to assess the accuracy of calculation results by stability analysis codes using the experimental data, and (3) to establish the stability solution methodology, in which the selected control rod insertion (SRI) system is installed to automatically exclude the operated core from possibly unstable core condition.

The present paper describes the BWR stability issues in Japan. Researches related to the phenomena identification, models, and codes applicable to the design analysis and stability solution methodologies are described. Authors suppose that understanding the basis of the BWR stability issues can be useful for future improvements in the BWR stability solution methodology based on the advanced analysis models and codes. In the last section of the present paper, an outline of the on-going research on the advanced BWR stability solution methodology is to be introduced, which employs

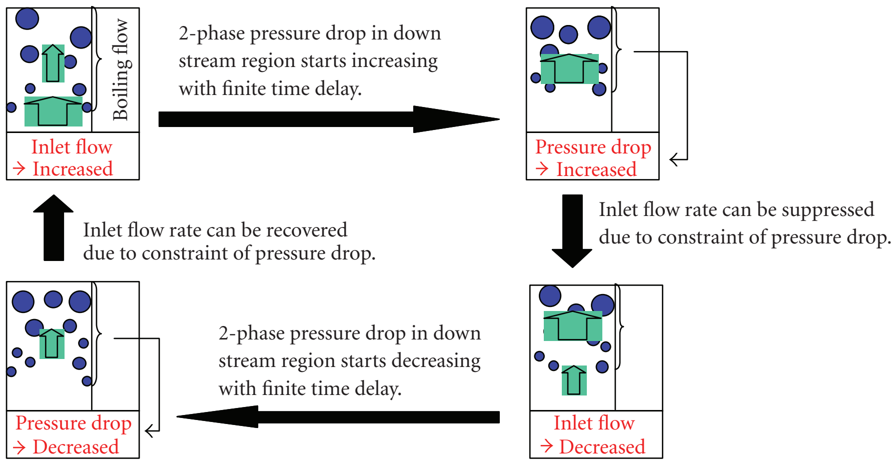


Figure 1: Schematic description for channel instability mechanism.

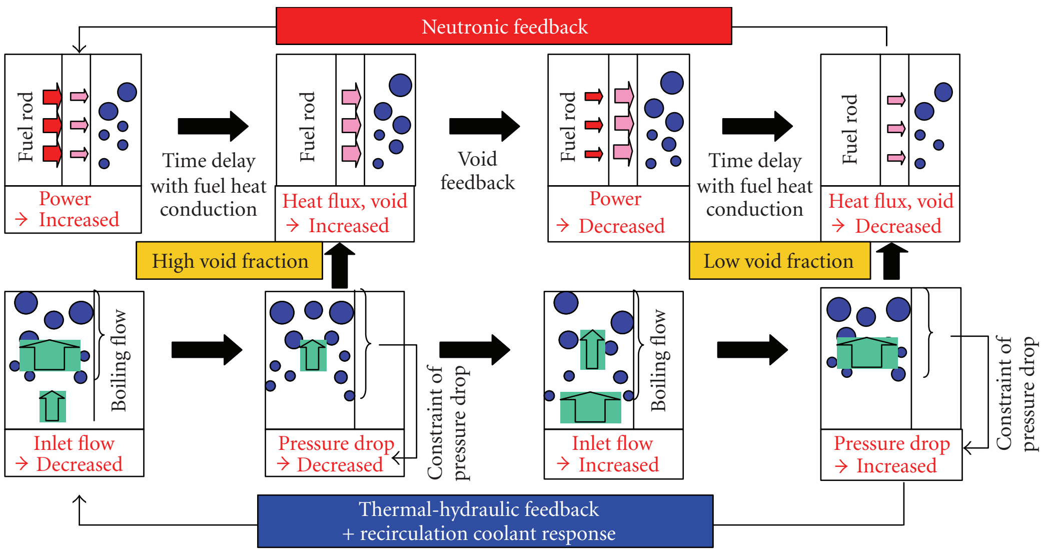


Figure 2: Schematic description for core instability mechanism.

Table 1: Features in frequency-domain and time-domain stability analysis codes.

	Frequency-domain code (reduced-order model)	Time-domain code (3D kinetics model)
Computation speed	Fast	Slower than Freq. model
Numerical diffusion	No	Dependent on numerical scheme
Decay ratio	Determinable by the unique way from the Nyquist curve for the system transfer function	Sensitive to time-step size, disturbance condition to activate transient state, in numerical simulation
Model limitation	1st-order linear perturbation to the nonlinear physical systems	Basically No
Spatial behavior	No	Yes
Nonlinear behavior	No	Yes

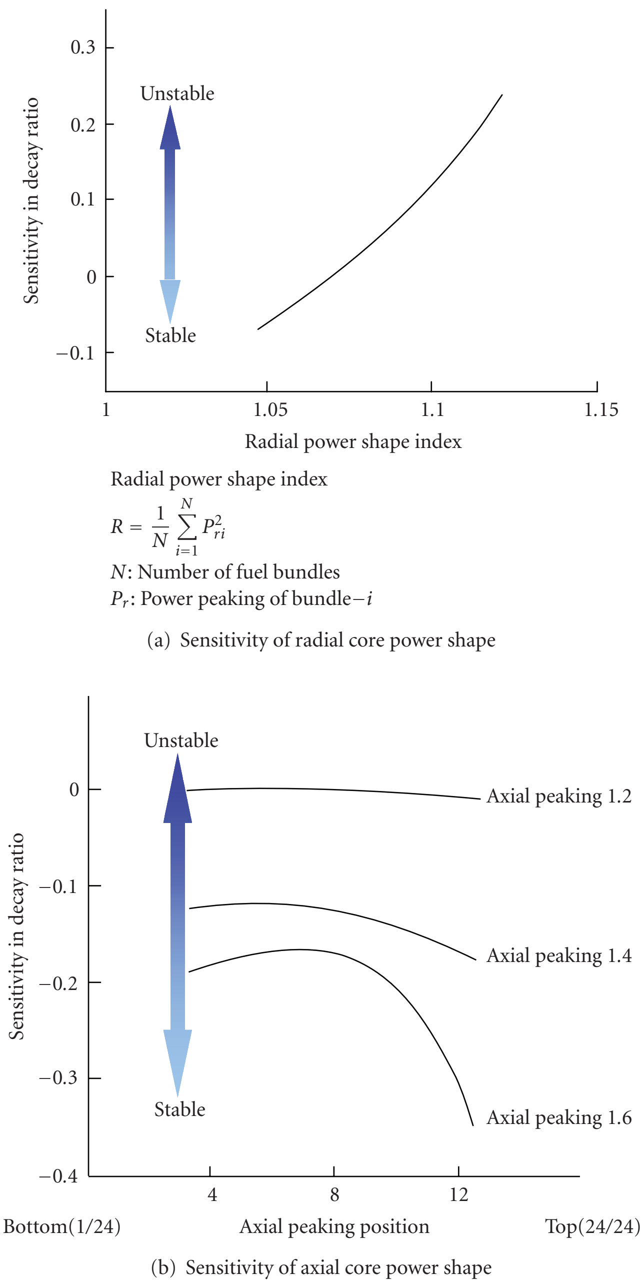


Figure 3: Sensitivity of core power shape to core stability decay ratio.

the best-estimate analysis code and the statistical approach in the safety evaluation methodology.

2. BWR INSTABILITIES

The BWR instability can be subcategorized into the three phenomena: (1) channel instability (density wave oscillation); (2) core instability (global core power oscillation); and (3) regional instability (powers in two halves of a core oscillate with an out-of-phase mode).

2.1. *Channel instability*

The channel instability is equivalent to the coolant density wave oscillation in a boiling channel, where the channel pressure drop is kept constant by any constraint [2, 3]. As shown in Figure 1, the coolant void sweeps in the boiling region, which significantly affects the 2-phase pressure drop, consequently leads to the coolant mass flow oscillation at the channel inlet. Hence, the channel instability can be invoked in a channel, where the 2-phase pressure drop is relatively larger than the single-phase pressure drop, for such conditions as (1) higher channel power and lower flow rate, (2) lower inlet coolant subcooling, (3) down-skewed axial power shape, (4) numbers of fuel rods and of fuel spacers which tend to generate the larger pressure drop in the 2-phase boiling region. In general, however, excitation of the channel instability can be suppressed by many other stable channels via the neutronic coupling effect among fuel bundles in an actual core.

2.2. *Core instability*

The coupled neutronic and thermal-hydraulic power oscillation can be categorized into the global instability and into the regional instability. In the first mode, the global core power oscillates in-phase, while in the regional oscillating mode, the power in a half core oscillates in an out-of-phase mode with respect to the other half. The core power oscillation is mainly driven by the negative coolant void feedback with the finite time del ay due to the fuel heat conduction [2]. This power oscillation can be actually excited by synchronizing with the mentioned density wave oscillation, as schematically described in Figure 2. a range from 0.3 to 0.6 Hz [4, 5], which are correlated with the wave propagation velocity through the core fuel channel.

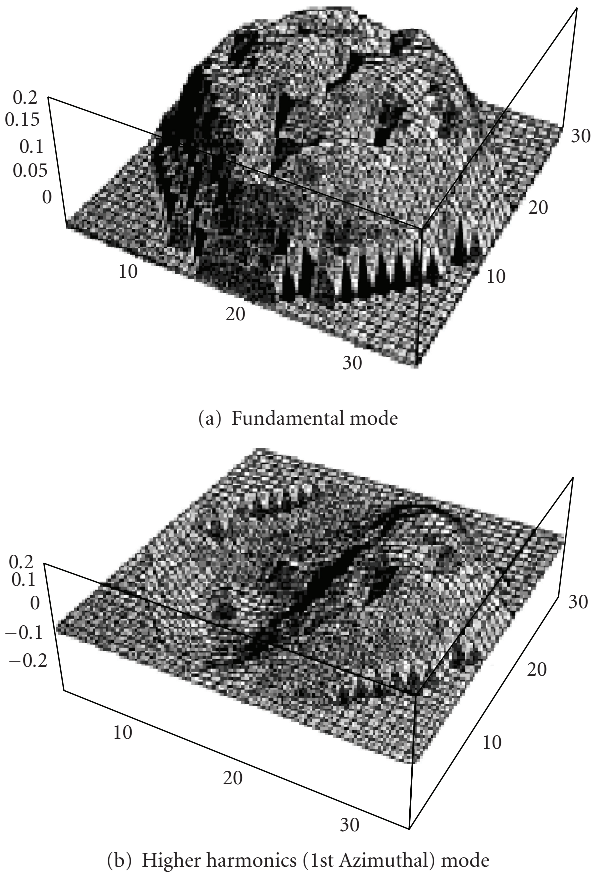
The core power oscillation becomes unstable under the lower flow and higher power core operation condition, corresponding to the density wave oscillation behavior. Large negative void feedback and faster fuel heat conduction make the core state unstable. In addition, the past investigation using frequency-domain stability analysis codes revealed interesting sensitivity with respect to the core power distribution, as shown in Figure 3 [6]. As for the radial power shape, fuel bundles with high power peaking factors tend to reduce the channel stability in the entire core, resulting in the core instability. The sensitivity regarding the axial power shape has more complicated nature as described below. The down-skewed shape leads to the longer boiling length, which makes the frequency of the density wave oscillation greater than the time constant in the fuel heat conduction. This mismatch tends to result in the stable core power oscillation. On the other hand, the flat and/or the middle-skewed shapes make the greater influence of neutronics in the high void region of the core, inducing the core instability due to increase in the negative void feedback.

2.3. *Regional instability*

The basic phenomenon dominating the regional instability is similar to that for the core instability, and the coupled

Table 2: Three-dimensional stability analysis code in Japan.

Code	Development	Code qualifications
TOSDYN	Toshiba	Peach Bottom-2, Vermont Yankee, LaSalle-2, Caorso, Leibstadt, KRB-B/C, BWRS Startup Data, parallel loop channel stability test
STANDY	Hitachi Toshiba BWR utilities in Japan	Peach Bottom-2, Vermont Yankee, LaSalle-2, Caorso, KRB-B/C, etc.
TRACG	General electric GNF-A/GNF-J Toshiba Hitachi	LaSalle-2, Cofrentes, Leibstadt, Forsmark-1, etc.
DYNAS-2	Nuclear fuel industries	PB-2, WNP-2, KRB-B/C, KKK, Ringhals-1, parallel loop channel stability test, etc.
TRAC-BF1/ENTRÉE	TEPCO systems	Ringhals-1



(a) Fundamental mode

(b) Higher harmonics (1st Azimuthal) mode

Figure 4: Sample of spatial neutron harmonics modes.

neutronic and thermal-hydraulic oscillation can be individually excited in two halves of a core with an out-of-phase mode. Previous researchers proposed that the regional instability is equivalent to the oscillation of the higher harmonics (1st azimuthal mode) of the neutron flux distribution, while the core instability is to the oscillation of the fundamental mode (see Figure 4) [7]. Hashimoto derived the so-called ‘modal point neutron kinetics equations in order to analytically represent the phenomenon, in stead of the ordinary point kinetics equations [8]:

$$\frac{dN_m(t)}{dt} = \frac{\rho_m^s - \beta}{\Lambda_m} N_m(t) + \frac{\rho_{m0}(t)}{\Lambda_m} N_0 + \sum_{n=0}^{\infty} \frac{\rho_{mn}(t)}{\Lambda_m} N_n(t) + \lambda c_m(t), \tag{1}$$

$$\frac{dc_m(t)}{dt} = \frac{\beta}{\Lambda_m} N_m(t) - \lambda c_m(t), \tag{2}$$

where

$$\rho_m^s = 1 - 1/k_m, \tag{3}$$

$$\rho_{mn} = \langle \phi_m^*, (\delta M - \delta L)\phi_n \rangle / \langle \phi_m^*, M_0 \phi_m \rangle. \tag{4}$$

m is the order of the higher harmonic mode ($m = 1, 2, \dots$); N, c, and β are the core-averaged neutron flux, del ayed neutron precursor, and delayed neutron fraction, respectively. The other variables and notations are defined in the original paper [8]. Physically, ρ_m^s represents the subcriticality of the mth harmonic mode, which is mathematically corresponding to the eigenvalue separation, and is a negative value in the above definition. Hashimoto [8] and Takeuchi et al. [9] pointed out that a smaller absolute value of the subcriticality makes the feedback gain of the regional oscillation larger, which is correlated to the first term of the right-hand side of (1), inducing the regional instability.

As mentioned above, powers in two halves of a core oscillate with an out-of-phase mode, therefore, significant oscillations cannot be observed in the core-averaged power and inlet coolant flow responses. This results in that the hydraulic flow response via the recirculation loop is less sensitive to the regional stability.

3. BWR STABILITY ANALYSIS CODES, VERIFICATIONS, AND APPLICATIONS

Several stability analysis codes have been developed so as to investigate the BWR instability phenomena in detail, and to apply on the BWR core design in Japan. The analysis codes can be mainly classified into the two categories, the frequency-domain code and the time-domain code. Features

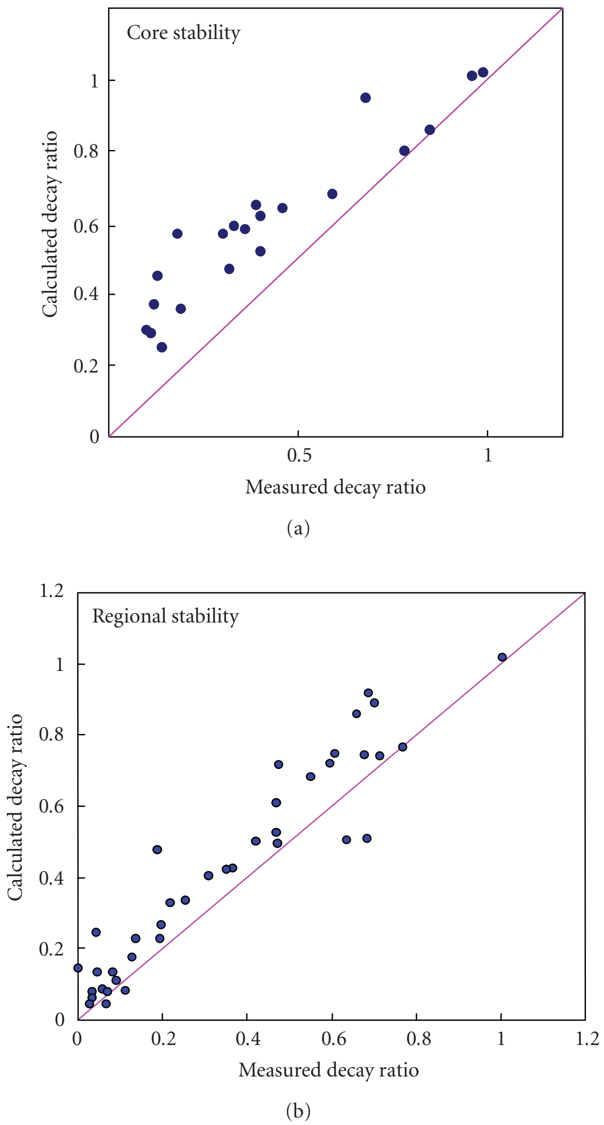


FIGURE 5: Sample of verification for the frequency-domain code using stability test data.

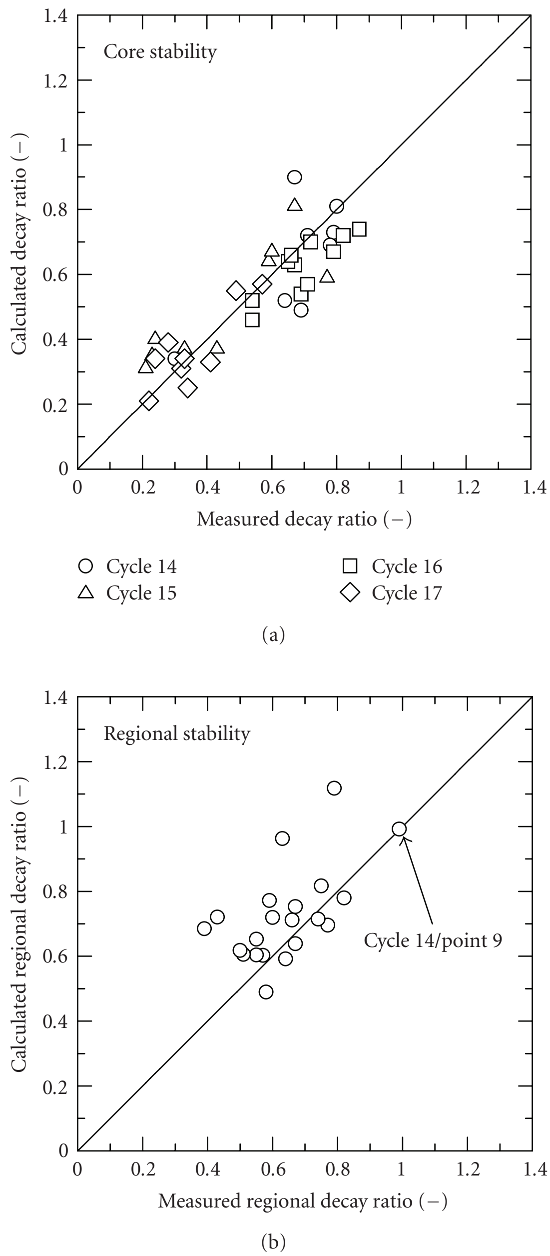


FIGURE 6: Sample of verification for the time-domain code using Ringhals-1 stability test data.

of the frequency-domain stability analysis code and of the time-domain code are summarized in Table 1, respectively.

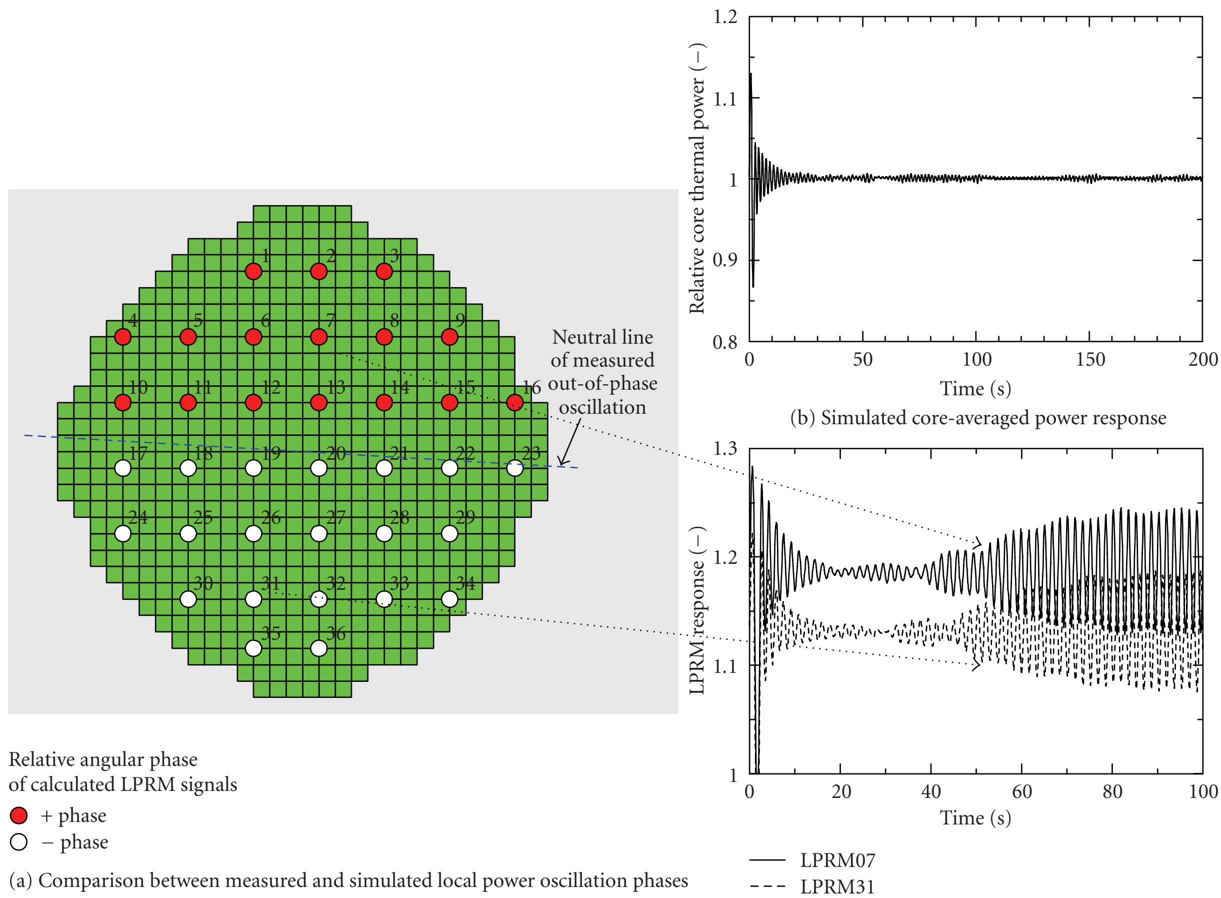
3.1. Reduced-order frequency-domain codes

In general, the frequency-domain code employs the reduced-order model like the point neutron kinetics, to mathematically simplify the phenomenological representation, and to attain the faster computation time. In addition, the decay ratio, representing the stability degree of an oscillation, is determinable by the unique methodology based on the system transfer functions. These features are favorable in the design analysis. All the equations representing the physical phenomena are linearized for small perturbations to yield the system transfer functions via the Laplace transformation, which characterize the channel, core, and regional stabilities.

The primary physical equations employed in a representative frequency-domain code are the followings:

(1) mass, energy, and momentum equations for 2-phase mixture boiling coolant flow;
(2) radial one-dimensional fuel heat conduction equations; and
(3) point neutron kinetics equations.

The thermal-hydraulic behavior in a core is modeled with the parallel channel geometry, and the fuel heat conduction is accounted in each hydraulic calculation node. As for the regional stability analysis, the point neutron kinetics equation

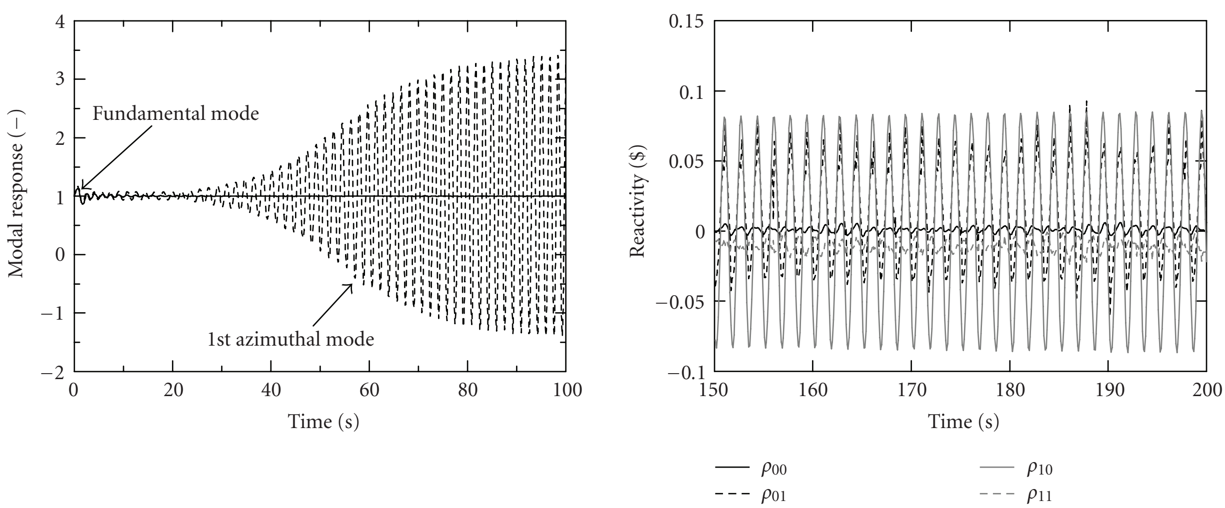


(a) Comparison between measured and simulated local power oscillation phases

(b) Simulated core-averaged power response

(c) Simulated local power responses

Figure 7: Simulated regional instability at Ringhals-1 C14/PT9 stability test.



(a) Simulated modal responses

(b) Simulated modal reactivity responses

Figure 8: Modal parameter responses under simulated Ringhals-1 C14/PT9 regional instability.

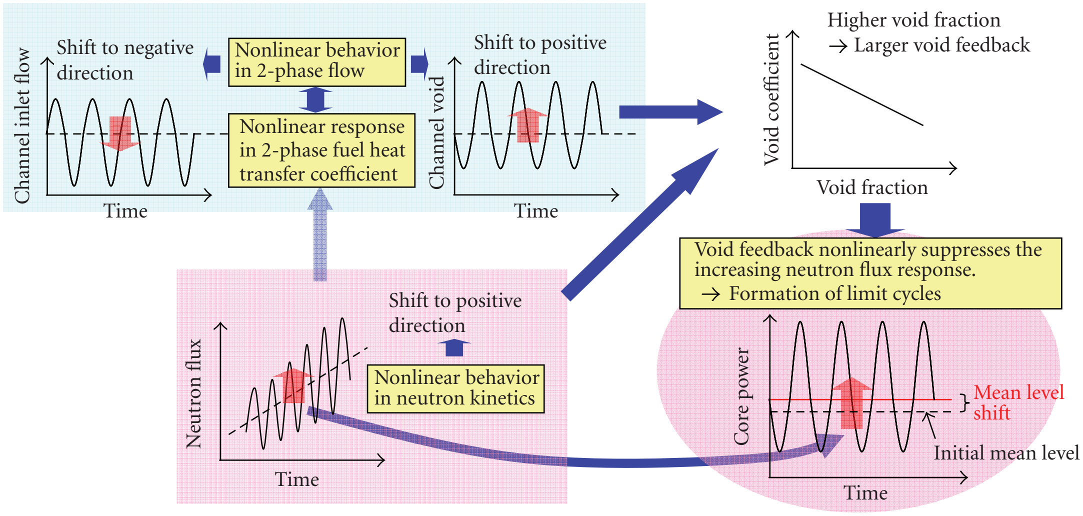


Figure 9: Mechanism in formation of limit cycle oscillations.

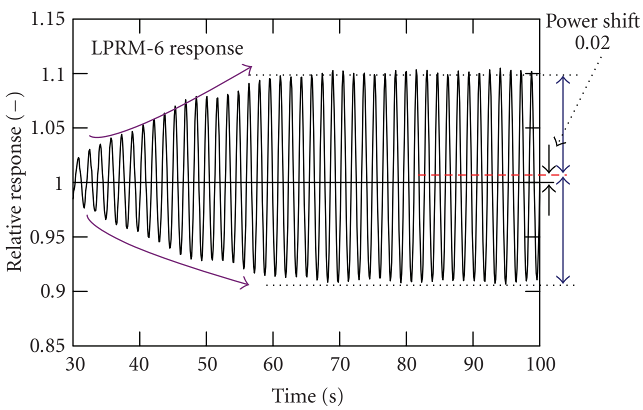


Figure 10: Average power shift in simulated Ringhals-1 C14/PT9 regional instability.

is replaced by the modal point kinetics equation as mentioned in the previous section.

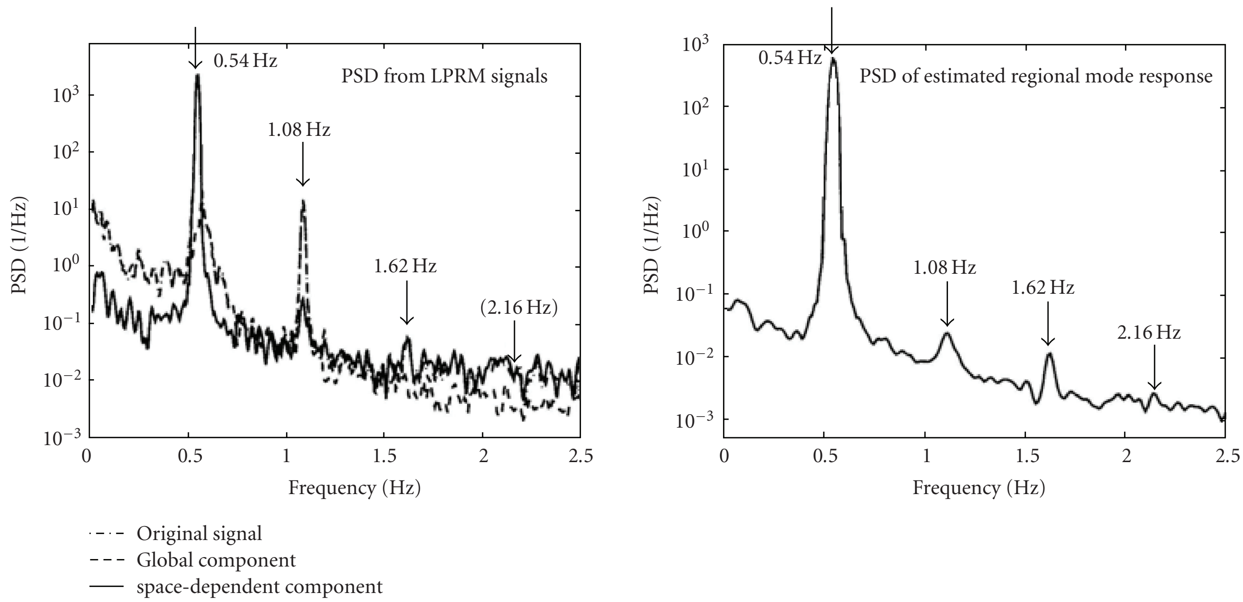
Figure 5 shows a sample of verification result for the frequency-domain code, which is currently applied on the BWR core design analysis. The code is able to derive good correlations over the wide stability range for the core stability analysis as well as for the regional stability analysis, while the code models are conservative a priori.

3.2. *Three-dimensional time-domain codes*

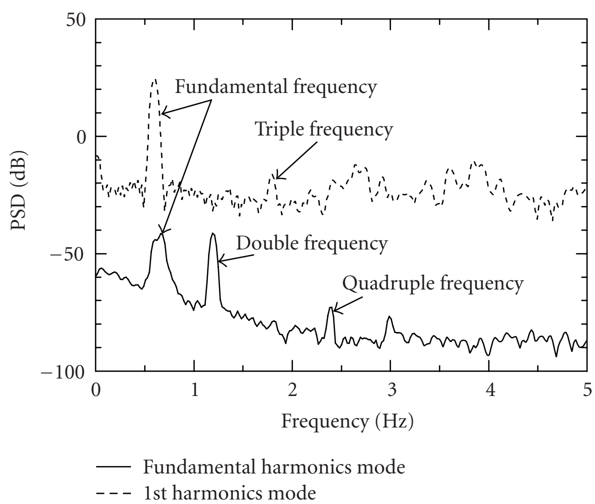
As described above, the frequency-domain codes generally employ the simpler fundamental equation set in order to avoid mathematical difficulties in derivation of the system transfer functions representing the coupled neutronic and thermal-hydraulic phenomena in a BWR. The time-domain code, on the other hand, adopts the more sophisticated physical models, like the spatial neutron kinetics model. In fact, their implementation on a code is simple and straightforward, while it consumes larger computational time than the mentioned reduced-order model. Since 1990s, however, the significant advance in computation technologies has facilitated development of time-domain codes that employ the complicated three-dimensional and multigroup neutron diffusion kinetics model [1, 10–14]. Several time-domain codes developed by Japanese organizations are listed in Table 2. The most significant advantage in these codes is that the detailed spatial kinetics behavior in a core can be explicitly simulated, namely, both the core stability and the regional stability can be evaluated using a single three-dimensional time-domain code without any modification. However, users have to pay attention to the applied numerical time step size, which is sensitive to the simulated oscillation and decay ratio [15, 16].

Furthermore, a simulator has been implemented on the recent time-domain codes in order to accomplish the more realistic dynamic simulation reflecting the actual core state including the fuel history data thus being seamlessly consistent to the static core design [17, 18]. Figure 6 shows a sample of verification for the three-dimensional time-domain stability analysis code, SIMULATE-Kinetics, using the Ringhals-1 stability test data [19, 20]. It can be confirmed that the code is basically targeting on the best-estimate stability analysis on the contrary to conservative approach applied in the frequency-domain code. The Ringhals-1 cycle-14 PT9 stability test, where a regional instability was observed, was accurately simulated as shown in Figure 7 [19]. In addition, the results of numerical simulation demonstrated that the observed regional instability is equivalent to an oscillation of the higher harmonics mode (1st azimuthal, N_1 defined by (3), and that modal reactivities (ρ_{10} and ρ_{01} defined by (4) are dominant in the regional event as shown in Figure 8.

A feature of the three-dimensional time-domain code is that it is applicable to the analytical investigation of the limit cycle oscillation which is driven by the complicated nonlinear effects [21–23]. Figure 9 schematically describes the del iberated mechanism in the formation of limit cycle oscillation.



(a) Harmonics excitation in measured modal responses [24]



(b) Harmonics excitations in simulated modal responses

FIGURE 11: Harmonics excitations under regional limit cycle oscillations.

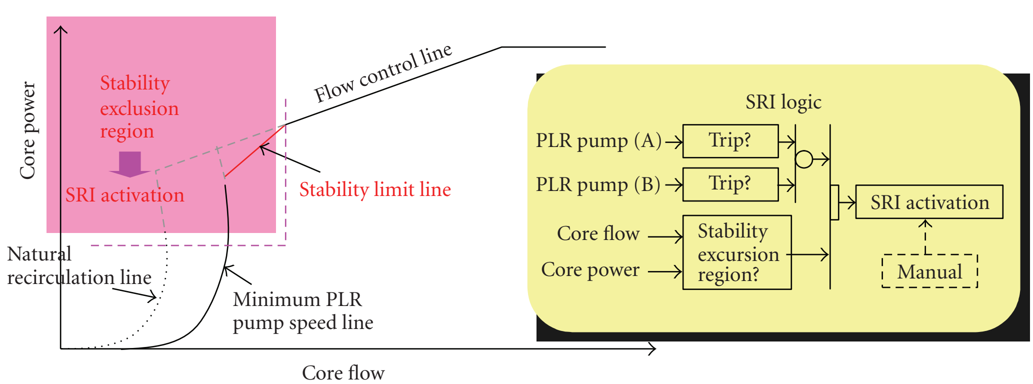


FIGURE 12: Outline of approved stability solution methodology in Japan.

The nonlinear behavior in the 2-phase boiling flow tend to increase the core-averaged void fraction and the negative void feedback, which suppresses the growing neutron flux oscillation due to nonlinearity in the neutron kinetics, resulting in the limit cycles. Any power shift observed in the measured core power responses and/or in the numerically simulated power responses (see Figure 10) is due to the above nonlinearities.

As for another scientific interest on the regional limit cycle oscillation, the bifurcation behavior observed via the spectrum analysis of the measured core power responses [24], Farawila theoretically proposed that the nonlinear interaction in the modal reactivities defined by (4) plays an important role in this phenomenon [25]. In addition, Ikeda et al. have numerically demonstrated that the nonlinearity excites the different higher harmonics of the core-averaged and regional power responses, respectively, as shown in Figure 11 [23], which was obtained by applying a spectrum analysis to the simulated fundamental and higher modal responses (refer to Figure 8).

4. CURRENT BWR STABILITY SOLUTION METHODOLOGY

Since the instability incident at LaSalle-2 [26], GE, and US-BWROG has developed several long-term stability solution methodologies [27, 28]. Also in Japan, a similar stability solution methodology was established, where the adequate stability margin must be ensured in the core design process, and the selected control rod insertion (SRI) system is equipped to exclude the BWR core from the unstable operation region (stability exclusion region) as shown in Figure 12 [29].The SRI system is activated to suppress the core power when the core coolant recirculation pumps are tripped and the core goes into the preliminary determined stability exclusion region. The stability exclusion is to be determined by using stability design codes certified via the regulatory assessments, with the conservative stability criteria (decay ratio is less than 0.8). Consequently, this methodology is targeting on that the BWR instabilities are not possible in the operated core in Japan.

5. RESEARCH ON ADVANCED BWR STABILITY SOLUTION METHODOLOGY

The current stability solution methodology is effectively contributing to safety of BWR plant operations in Japan. However, considering the recent occurrences of BWR instabilities [30, 31], authors suppose that any improvement may be indispensable for the future stability solution methodology, which is able to correspond to the recent modifications in the existent BWR plants as the extended core thermal power-uprate [32] with the advanced fuel designs [33–35]. An approach to resolve this concern is that sufficient stability margin is to be introduced, namely, the plant operable region is limited by the wider stability exclusion region, which can be determined by using the current conservative stability analysis code, as shown in Figure 13. This approach, however, possibly leads to the economical loss by consuming longer time for the plant startup operation. Reactors with the larger stability exclusion region are generally allowed to adopt few continuous withdrawals of control rods under the lower power condition. This is because the continuous-withdrawal operation induces significant increase in core power at the fixed core flow condition, possibly removing the core into the prohibited stability exclusion region. Therefore, a lot of control rod operations, which must be conducted slowly and intermittently to maintain the fuel mechanical integrity, are required under the higher power condition to attain the target control rod pattern at the rated power operation. Consequently, the overall plant startup time tends to become longer in the BWR plant with the larger stability exclusion region.

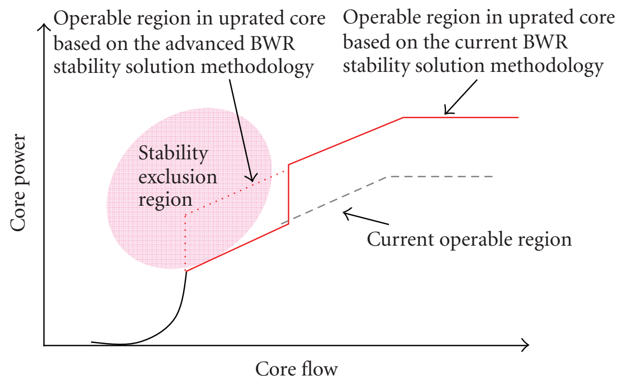
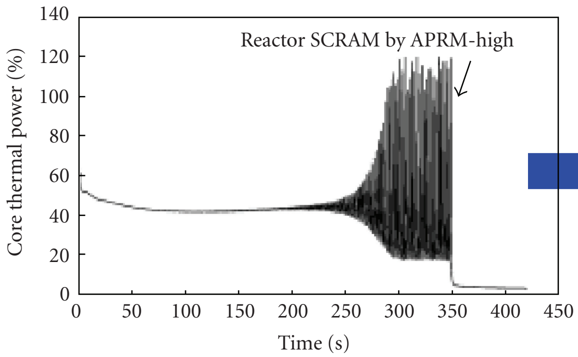
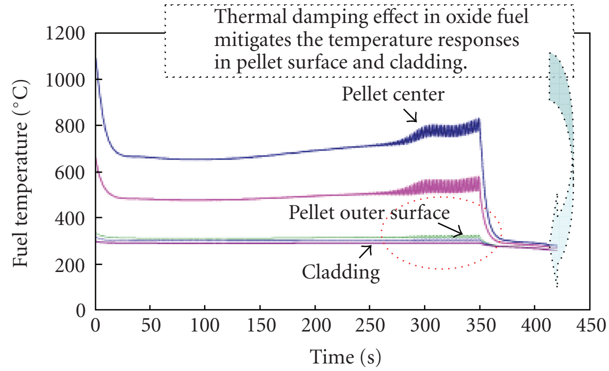


Figure 13: Plant operable region for the power-uprated core.

In order to reasonably enhance the operable region even under the power-uprated core, a joint research group organized by several Japanese industrial and academic organizations has started a development of an advanced stability solution methodology based on the best-estimate code system [36]. Basis of the present research is to apply the original regulatory criterion with respect to the BWR instability [37], that is, "*exceeding specified acceptable fuel design limits (SAFDLs) are not possible*", not prevention from the instability occurrence. From the viewpoint of the applicable SAFDLs on the BWR instability, the PCMI and the material fatigue via the power oscillation possibly make no significant affect on the fuel integrity, because temperature responses of the fuel pellet and cladding are negligibly small as shown in Figure 14. Therefore, occurrence of the core coolant boiling transition (BT) can be a primary cause for the fuel failure under the BWR instabilities. So as to accurately and mechanistically predict the BT onset even under the BWR instabilities, the research group is applying an advanced code system based on the best-estimate plant simulator, TRAC-BF1/ENTRÉE [13], and the 2-fluid/3-field subchannel code, NASCA [38]. As schematically described in Figure 15, TRAC-BF1/ENTRÉE provides the pin-by-pin-based power responses in each fuel bundle; the subchannel thermal-hydraulic behavior and BT onset on the local rods are evaluated by NASCA with the boundary conditions supplied by the TRAC-BF1/ENTRÉE.



(a) Postulated BWR power oscillation



(b) Fuel temperature responses

FIGURE 14: Fuel temperature responses under the representative BWR instability.

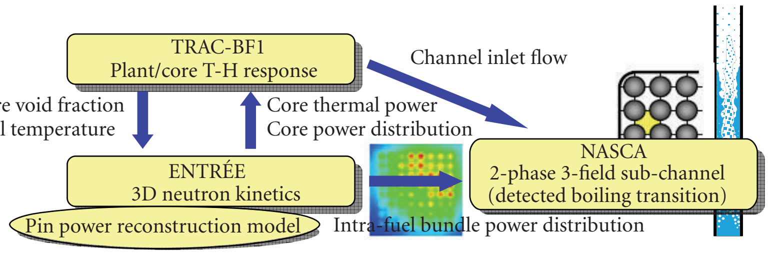


FIGURE 15: Outline of TRAC-BF1/ENTRÉE and NASCA code system.

The research group is also investigating the possibilities to introduce the statistical safety evaluation methodology [39] so as to establish the reasonable conservatism in the stability exclusion region determined by using the above best-estimate code system. The research, in particular, currently pays a lot of efforts to establish the phenomena identification ranking table (PIRT) applicable to BWR instabilities including the subchannel thermal hydraulics, based on the existent stability PIRTs [40–42]. This is the basis of the uncertainty evaluation for the best-estimate BWR stability analysis.

6. CONCLUSIONS

Many efforts have been paid to research on BWR stability issues in Japan, as introduced in the present paper. The industrial organizations have developed and improved the BWR stability analysis using computational tools specific for the reduced-order frequency-domain and three-dimensional time-domain codes. The first category is currently applied to the BWR stability design analysis, while the latter one has been exploited to understand the complicated phenomena related to BWR stability. The current stability solution methodology based on the SRI system with the stability exclusion region is successfully preventing the occurrence of BWR instabilities in Japan. However, authors suppose that the future application of the extended core power uprate requires further improvements to the current solution methodology in order to reasonably minimize the stability exclusion region. A Japanese research group is currently proposing to apply the best-estimate analysis code with the statistical safety evaluation methodology. This will allow better evaluation of the stability exclusion region, and will be consequently applied to the BWR plants with the extended core power uprate.

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Research Article

Thermal-Hydraulic Analysis Tasks for ANAV NPPs in Support of Plant Operation and Control

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Thermal-hydraulic analysis tasks aimed at supporting plant operation and control of nuclear power plants are an important issue for the Asociación Nuclear Ascó-Vandellòs (ANAV). ANAV is the consortium that runs the Ascó power plants (2 units) and the Vandellòs-II power plant. The reactors are Westinghouse-design, 3-loop PWRs with an approximate electrical power of 1000 MW. The Technical University of Catalonia (UPC) thermal-hydraulic analysis team has jointly worked together with ANAV engineers at different levels in the analysis and improvement of these reactors. This article is an illustration of the usefulness of computational analysis for operational support. The contents presented were operational between 1985 and 2001 and subsequently changed slightly following various organizational adjustments. The paper has two different parts. In the first part, it describes the specific aspects of thermal-hydraulic analysis tasks related to operation and control and, in the second part, it briefly presents the results of three examples of analyses that were performed. All the presented examples are related to actual situations in which the scenarios were studied by analysts using thermal-hydraulic codes and prepared nodalizations. The paper also includes a qualitative evaluation of the benefits obtained by ANAV through thermal-hydraulic analyses aimed at supporting operation and plant control.

1. INTRODUCTION

The main goal of this paper is to illustrate the usefulness of computational analysis aimed at supporting and improving plant operation and control of nuclear power stations. The contents presented below were operational between 1985 and 2001 at the Spanish NPPs run by the Asociación Nuclear Ascó-Vandellòs (ANAV). These power plants are the Ascó reactors (2 units) and the Vandellòs-II power plant. They all are Westinghouse design, 3-loop PWRs with an approximate electrical power of 1000 MW. The Technical University of Catalonia (UPC) thermal-hydraulic analysis team worked together with ANAV analysts for an important part of the mentioned period. Although the joint work was fruitful in both innovative engineering and research, the scope of activities currently presented belongs mainly to the former.

The steps that will be followed in presenting the work are

(i) description of specific aspects of thermal-hydraulic analysis tasks related to operation and control,
(ii) brief presentation of the results of some of the analyses performed,
(iii) qualitative evaluation of the benefits obtained through such analyses.

The responsibilities of analysts involved in supporting plant operation are somewhat different from those of other analysts that currently produce studies usually found in the technical literature. There could be some coincidences but usually their tasks are quite specific. Support tasks for commercial plants are something alive, and change depending on organizational requirements, status of the plant, and availability of external help. For this reason, there is some subjectivity in what follows below. The almost continuous organizational change of the engineering teams working on operation support corroborates what has been said above. The statement that really defines the function of the operation support analyst is that he/she shares objectives with the engineering team that assists plant operation by means of engineering studies and decision-making. On some occasions,

Table 1: Elements in Ascó and Vandellòs-II models.

Elements	Ascó	Vandellòs-II
Hydrodynamic volumes	549	613
Control variables	1454	1327
Variable trips	219	234
Logical trips	431	461
Tables	241	227
Interactive variables	117	142

the operation support analyst performs a hundred per cent of the necessary studies. However, quite often, he or she looks through the problem, does a few calculations, draws some preliminary conclusions, and then subcontracts the final analysis to an appropriate engineering company. Subcontracting involves a technical follow-up by the plant analyst, who performs a detailed review that usually includes calculations devoted to check the consistency of subcontractor's results.

In spite of being somewhat subjective, such practises have been operational for quite a long period and have produced interesting results as regards utility engineering and safety issues [1]. Many areas of industry follow this approach. Companies and organizations have certain analytical capabilities and a great knowledge of problems that come up, but they use external help when there are productivity requirements. The examples presented are calculations performed along these lines.

Most of the analyses presented have been performed using integral plant models. These models are best estimate (BE) models and are intended to produce a realistic prediction of the studied scenario. In the case of Ascó and Vandellòs-II NPPs, as in the case of other PWRs, they were prepared long ago [2] using codes such as RELAP5.

Preparing an integral plant model is both a meticulous and laborious task, in which each hydrodynamic system, heat structure, protection and control systems, and the core itself are developed individually, starting from the appropriate design information. Table 1 gives an idea of the degree of detail for each of the models.

Figure 1 shows the main nodalization diagram of the Ascó plant. Both models include another 4 diagrams representing safety injection systems, steam lines, main and auxiliary feed-water (Figure 2), and detailed diagrams of vessel, pressurizer and steam generators (SGs). Figure 8 shows an example of a logic diagram implemented in one of the models. The number of control systems included with a certain degree of complexity, as in the case of Figure 8, is approximately 30 in each model.

The model preparation was made compatible with its use in operation support. Once the models were used, they produced the necessary feedback to improve their performance [3, 4]. The final result of this development constituted a complete product whose features will be commented below.

The aim of these BE models is to produce a realistic picture of the NPP behavior which will be useful for different kinds of decision-making. The BE models have improved their predictive capacity to a great extent; their results have changed from being a good guidance for general understanding of dynamics to being extremely reliable. If they come together with certain methodology requirements, they might also be valid for licensing and management of margins.

The integral plant models are useful tools for the analysis of dynamic behavior whenever certain requirements are fulfilled. Some of these are related to qualification and documentation [5] of the models and others are linked to case analysis. The most important requirement is the analyst's professional profile [6]. The analyst needs to have the skills to guaranty that

(i) nodalizations have been properly set and adjusted,
(ii) the right options have been activated,
(iii) correct assumptions have been made,
(iv) boundary conditions are those that are needed for the problem.

It is usually said that to produce good results, one needs a qualified user using a qualified nodalization adapted to the problem by means of a qualified code [7]. Analyst training has always been a high-priority issue for ANAV.

2. ANALYSIS TASKS

Most of the analysis tasks mentioned in this section are extensively discussed in two different IAEA safety reports [22, 23]. These documents were developed based on broad international consensus and they describe types and rules for performing computational analyses devoted to both being built and operating plants.

The purpose of this section is not to describe every related task but to add some aspects that are specific of the functions of the analyst working in support of plant operation. Terminology and task descriptions used in this paper are those of the mentioned IAEA reports.

2.1. Dialogue with regulatory body and fuel designer

Dialogue backed up by calculation results has been used successfully for many safety issues that have been discussed between the licensee and the regulator or the fuel designer [8]. The BE prediction of a scenario helps communication on any engineering subject related to dynamic behavior. As stated in the introduction, in some cases, the analyst could perform a complete set of calculations aimed at obtaining licensing. He or she could use his or her own nodalization, follow a best estimate plus uncertainty (BEPU) methodology, and come up with results to be directly submitted to the safety authority. In many other cases, results are produced for dialogue [9, 17]. Sensitivity analysis is the most usual practice of providing support for such communication. It shows clear advantages in relation to engineering judgement based on calculations performed using conservative assumptions.

Going into the technique of sensitivity studies in greater depth, the analyst often ends up sweeping the whole range of a definite parameter and analyzing the impact or the consequences on the final calculation results. In this way, reasonable doubts of real values are clarified.

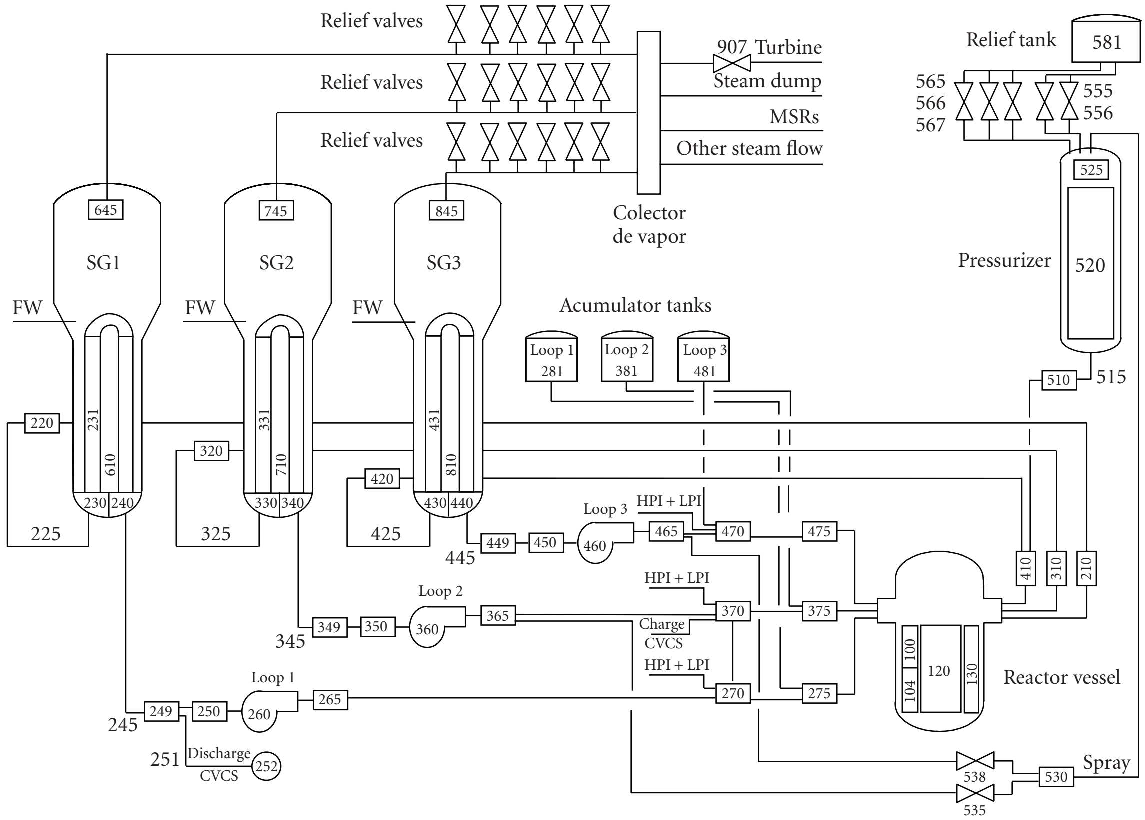

Figure 1: Main nodalization diagram of the Ascó plant.

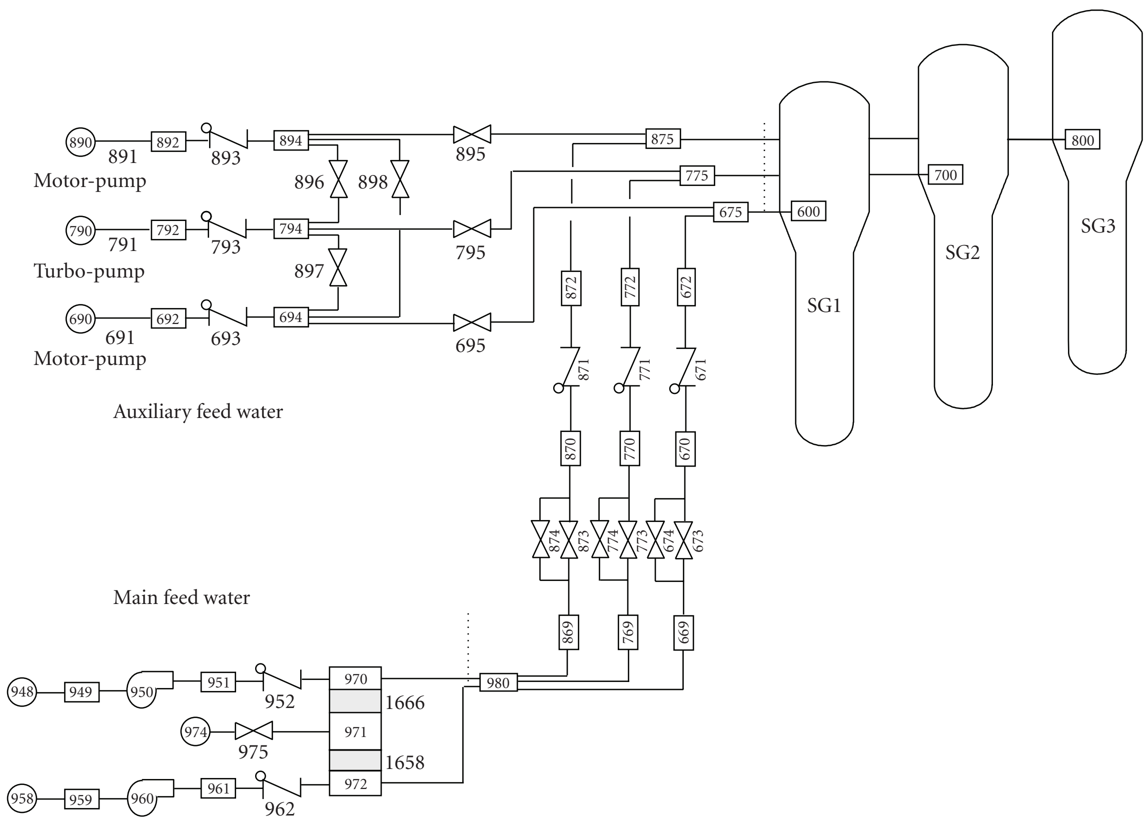

Figure 2: Nodalization diagram of the Ascó FW system.

2.2. *Thermal-hydraulic analysis of probabilistic safety assessment (PSA) sequences*

The thermal-hydraulic analysis of PSA sequences, mainly those of level 1, is normally performed using integral BE plant models [10, 11]. These are a kind of studies in which the IAEA rules given in [22] are normally followed and no additional comments are needed.

2.3. *Analysis of actual transients*

The simulation of actual transients usually produces in-depth knowledge of their dynamic behavior. It is also helpful to investigate and to determine the cause-effect relationships of the occurred transient [12–14]. This type of analyses must be performed with high relevance transients or following the request of engineering groups interested in the behavior of the systems involved.

One of the most powerful arguments in favor of these kinds of analysis is that they provide the possibility of generating time trends of functions and magnitudes that are not collected by plant instrumentation. Among these variables are mass flow at any junction (irrespective of whether there is an implemented instrument or not), magnitudes having values outside the instrument range, or other functions such as volumetric fractions of steam in two-phase mixtures.

The most usual objective of this type of analysis is to guarantee that design limits have been kept during the transient. Other objectives are

(i) to eventually clarify the abnormal behavior,

(ii) to answer technical questions,

(iii) to collaborate in follow-up actions (engineering, training, operation, and safety).

2.4. *NPP start-up tests analysis*

The predictive study of NPP start-up tests is extremely helpful for the test coordinator in order to avoid, as far as possible, mishaps, unexpected interactions, and delays that could give rise to economic losses [15]. Competitiveness goals of the electricity business have led the company running the plant to minimize the number of start-up tests to be performed. This kind of analysis helps to reduce the number of tests to only those that have proven benefits for both operation and safety. The expected benefit is usually either better knowledge of dynamic behavior or the correct performance of a system or instrument. This benefit could sometimes be proved by means of a calculation. It is clear that after the implementation of some important modifications to the plant, an extensive set of tests have to be carried out. However, on many other occasions some calculations properly performed could produce the necessary information.

Apart from these important activities related to start-up tests, standard post-test analyses could also become very significant. Important adjustments of the plant model arise very often from the studies carried out as post-test analyses.

2.5. *Analysis of hypothetical transients for operation support*

Emergency operating procedures (EOPs) validation analyses are the most important studies performed that belong to this group. In this case again, no additional remarks are needed as all definitions given in [23] are shared by ANAV-UPC team.

There is also another kind of studies related to operational procedures. They are usually performed at the request of the operation team and are aimed at clarifying any circumstances related to the involved scenario [16]. On many occasions, these studies are aimed at investigating NPP response to boundary conditions that have already actually occurred at another power station.

2.6. *Transient analysis for training support*

There are two different groups of studies in this field analyses for validation of plant training simulators and analyses devoted to direct training actions.

The former is defined in IAEA report [22] and is not needed of any clarification.

The latter is another type of training task that directly uses the results of thermal-hydraulic analyses. At some power stations, dissemination of results from dynamic calculations is organized with the aim of improving the general knowledge of engineers and members of the technical staff. These training tasks are usually assisted by tools that visualize the results of integral plant models. The produced images and animations enable an appealing dissemination of contents which, communicated otherwise, would not be quite so attractive [18]. Therefore, the combined tool (plant model plus visualizing tool) is an interesting support for different direct training actions.

2.7. *Design modifications*

Plant design modifications also need dynamic analyses. The goal of these studies is to establish the impact that modifications in components or systems have on the interactive global operation of the plant. Among these studies, those related to set-point adjustment, as well as those originated by important technological changes, are the most significant. Projects such as the replacement of SGs, the digitalization of the feed-water (FW) system control, or power upgrading have required prior developments of the model so that it could give quite a complete image of predicted plant behavior.

2.8. *Improvement of plant availability*

Integral plant models were prepared in the past to tackle safety issues and they continue being valid for these purposes. The wide use that they have had in all types of dynamic analysis of real or hypothetical plant behavior has made them a valid tool for the improvement of availability or to reduce the number of unnecessary reactor shutdowns. This capacity, usually implemented through control improvement, allows safety interests to be combined with those of productivity and competitiveness [13, 16].

3. PROCEDURE AND PROJECTS

In order to fulfil the above-mentioned tasks, usually devoted to the analysis of either transients that have actually occurred in the plant or hypothetical ones, some guidelines must be followed by the thermal-hydraulic analyst.

This section, as the previous one, is not a full description of procedures and projects. It adds some considerations on the most specific features involved. Documents like [7, 22] are giving guidelines and suggestions to carry out analysis tasks. They cover all aspects including preparing and maintaining models, performing calculations, and documenting results. Only two considerations need to be added in this case.

The first consideration that influences this analyst's work is the fact of being a member of an engineering team that supports plant operation. The analyst works in close contact with other engineers managing systems, equipment, licensing, reload planning, or quality assurance. As said in the introduction, the operation support analyst shares objectives with the engineering team that assists plant operation by means of engineering studies and decision-making.

The second consideration is related to ANAV-UPC model qualification procedure. This procedure shares basic aspects with many others and has some specific steps related to taking advantage of plant experiences for model qualification. It is described in [19] and currently followed by the team.

All of these tasks have been carried out in ANAV by two analysts (one per reactor as the 2 units of Ascó are almost identical) for the whole 1985–2001 period. External help from engineering companies was used when necessary. The most significant projects developed by the ANAV team with the aim of supporting plant operation are listed below in order to give some idea of the variety and depth of the analyses carried out.

(i) 1987–1989: analysis of incidental events at Ascó NPP [4, 12].
(ii) 1989: analytical support to the licensing of the AMSAC system (mitigation of transients without reactor trip) for Ascó NPP.
(iii) 1989-1990: thermal-hydraulic study of sequences for Ascó NPP PSA.
(iv) 1990: analysis of different alternatives to simulate the behavior of the Ascó secondary system.
(v) 1990-1991: analysis of the impact of SG tube plugging on the dynamic behavior of Ascó NPP.
(vi) 1991: analytical support for a "valve wide open test" of Ascó NPP turbine (in cooperation with Westinghouse).
(vii) 1992: analysis of Ascó NPP capabilities to face blackout scenarios.
(viii) 1993-1994: analytical support to the improvement of pressurizer level control at Ascó.
(ix) 1993: EOP's verification for Vandellòs-II NPP.
(x) 1994–1996: analytical support to SG substitution at Ascó NPP coordinated with the design team (consortium Siemens-Framatome) and the companies responsible for licensing (ENUSA and Westinghouse).
(xi) 1993-1994: analytical support to FW control system digitalization and improvement for Ascó NPP.
(xii) 1995: pretest analyses of start-up tests at Ascó NPP with the new SGs.
(xiii) 1996-1997: improvement of the design of the main FW control system at Vandellòs-II NPP.
(xiv) 1997-1998: implementation of RELAP5/MOD3.2-NPA for personal computers (collaboration with the engineering company PMSA) [18].
(xv) 1998: reanalysis of AMSAC behaviour after SG substitution at Ascó NPP [9].
(xvi) 1999-2000: advanced qualification, validation and documentation of thermal-hydraulic models at Ascó and Vandellòs-II NPPs [19].
(xvii) 1999-2000: analysis of the operating event occurring at the start-up test of cycle 13 at Ascó-II.
(xviii) 1999: thermal-hydraulic verification of interactive graphic simulators at Ascó and Vandellòs-II NPPs.
(xix) 1999–2001: update of thermal-hydraulic study of sequences for Ascó NPP probabilistic safety assessment.
(xx) 1999–2001: update of thermal-hydraulic study of sequences for Vandellòs-II NPP probabilistic safety assessment.
(xxi) 1998-1999: analytical support to Ascó NPP uprating.
(xxii) 1998-1999: analytical support to Vandellòs-II NPP uprating.
(xxiii) 2000-2001: analysis of the operating event occurring at Ascó-II on August 6th, 2000 [13].

Most of these analyses have produced reports that were subsequently dealt with either internally (by other ANAV branches such as PSA, licensing, fuel management, instrumentation and control, operation, and training) or externally (by the regulatory body, the fuel supplier or the engineering companies involved in developing associated subjects).

The three following sections are aimed at presenting some practical examples of this kind of analysis. The first is related to an actual transient (Section 4), the second to helping dialogue with the regulatory body (Section 5), and the third to an EOP/PSA transient (Section 6). The examples are presented in order to emphasize all the features that connect thermal-hydraulic analysis with operational concerns. More detailed information on the analysis can be found in specific calculation reports.

4. EXAMPLE OF AN ACTUAL TRANSIENT ANALYSIS: MAIN FW TURBO-PUMP TRIP WITH SCRAM CAUSED BY A HIGH-LEVEL SIGNAL IN AN SG

The selected transient took place in Unit 2 of Ascó NPP on August 6th 2000 [13]. The transient started with the trip of the main FW turbo-pump B when the plant was operating at steady-state nominal power. An automatic turbine run-back took place at 200% per minute until a load reference value

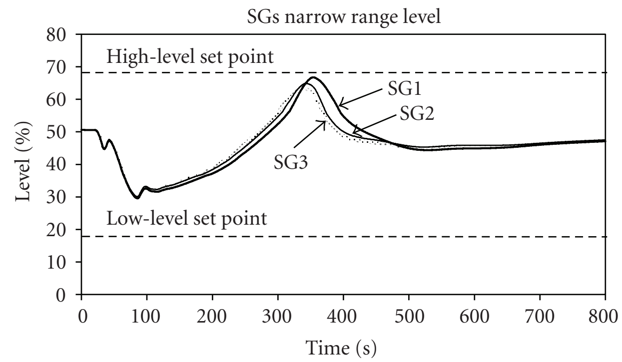

Figure 3: SGs narrow range level (FW turbo-pump generic transient).

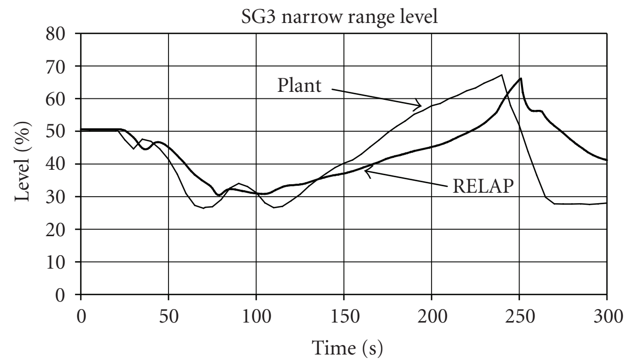

Figure 5: SG3 narrow range level (FW turbo-pump actual transient).

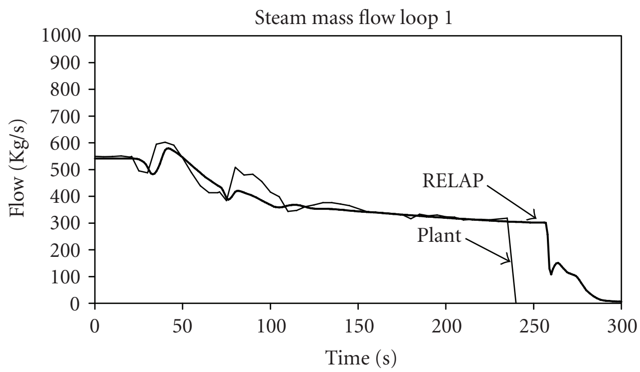

Figure 4: Steam mass flow loop1 (FW turbo-pump actual transient).

of 70% was reached. Both the steam-dump and rod control systems actuated in order to compensate the load rejection. At the same time, the main FW control system required an increase in the speed of turbo-pump A and the opening of FW valves to avoid a decrease of SG levels.

Subsequently, a first manual action was taken, consisting of an additional manual run-back of about 14%, with a new automatic steam-dump opening. SG levels decreased and reached their minimum values of 18%, 20%, and 26%.

At this point, a second manual action took place, as a rapid increase in level was noticed in all SGs (150 seconds). The operator then manually closed the main feed water valve in loop 3 by about 20%. Under these conditions, the level of SGs 1 and 2 increased quite quickly and the automatic control produced a closing signal for the related valves (200 seconds). The flow increased unexpectedly in loop 3, still under manual control, and its level subsequently rose until it produced a reactor trip due to a high-level signal (Figure 5).

The unexpected flow increase was the concern of the operation team. A main FW turbo-pump trip usually leads to a turbine run-back and to renewed stability at lower power which allows the scram to be avoided. The behavior of the plant seemed, a priori, abnormal.

To start with the analysis, the available technical information was studied. The first available block consisted of general information on the description of the systems involved. It included

(i) turbine run-back system description,
(ii) main FW system description,
(iii) types of run-back,
(iv) run-back due to main FW turbo-pump trip.

Some functional features were also analysed such as

(i) main FW system layout,
(ii) performance with only one FW turbo-pump/transition,
(iii) common FW header effect.

As the analyst belonged to the engineering team supporting plant operation, system characterization and functional aspects were easily identified by means of first-hand contact. The run-back system was fully implemented in the model. FW turbo-pumps were included using the RELAP5 "pump" component and characterized by all necessary mechanical parameters.

The effect of the common header was experienced in the plant. It consisted of the fact that the partial closure of any FW valve produces an increase in the pressure upstream (header pressure) and, consequently, an increase in the flow through the other valves. Although Figure 2 is only a diagram of FW system, it helps to understand the layout of the header and valves and the phenomenon itself.

Post-trip information, including the sequence of events and time-histories of the main variables, was available and helpful to assess the plant model. Time histories are easily converted to time graphs for discussion and for comparison (Figures 4 to 7 include such data). The package also includes operator reports, which are structured information produced following an established procedure that is applied immediately after the transient occurs. It comprises control room display values, parameters, and alarms and also a brief description of manual actions taken by the operators.

In this case, as can be seen in Figures 6 and 7, a sharp inflexion appears in the loop 1 FW mass flow graph (A), while two of them appear in loop 3 (B and C). Only B was easily identified as the manual closure of the loop 3 FW valve. It was suspected that A could have been an automatic action,

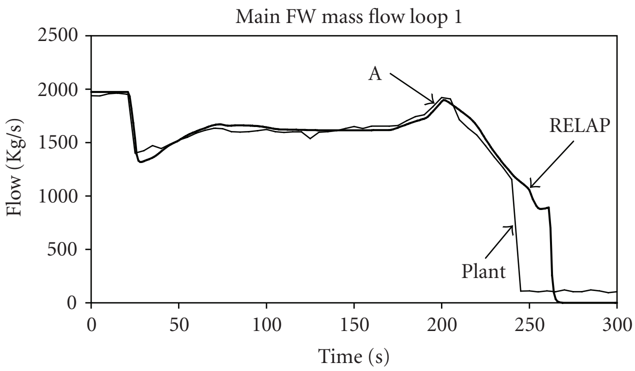

Figure 6: Main FW mass flow loop 1 (FW turbo-pump actual transient).

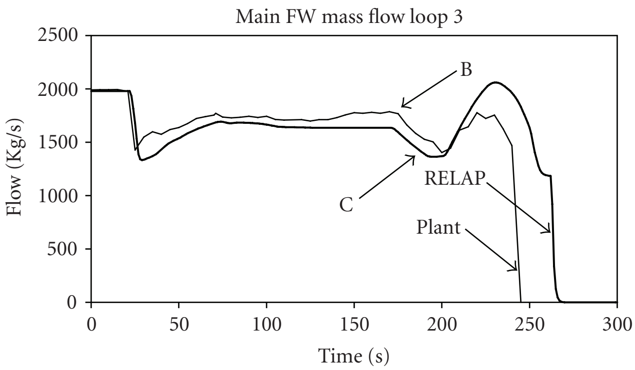

Figure 7: Main FW mass flow loop 3 (FW turbo-pump actual transient).

as the operator report made no reference to it. Event C was not identified at first by the operation team.

The subjects and questions that arose both internally (first) and externally (later on) by the safety authority were the following.

(1) Was the reactor scram really due to SG high level?
(2) The run-back system is designed to avoid reactor scram. Was it properly set up?
(3) Was the trip due to loop asymmetry?
(4) Was it due to the additional run-back?
(5) The FW closure did not seem to be related to a high-level signal.
(6) At least 3 sharp inflexions appear in the post-trip FW flow time histories. Detailed identification is needed.

In order to answer the above questions some calculations were performed:

(a) generic turbo-pump trip with actual geometry of the real plant,
(b) turbo-pump trip with additional manual run-back,
(c) turbo-pump trip with additional manual run-back and partial manual closure of FW 3.

Calculation (a) was performed with the turbo-pump trip as the only specific boundary condition. The nodalization was modified to take into account the geometry of actual FW pipes in the real plant (they are different depending on the loop they belong to). The conclusions of this first calculation were that although the subsequent level oscillation comes very close to the high-level set point (Figure 3), the run-back system seemed to be correctly designed since it avoids scram. In addition, asymmetry produced some deviation among maximum values of each loop, but not a great one. The calculation results help to answer questions (1), (2), and (3).

In calculation (b), an additional manual run-back at time 71 seconds, was simulated. For possible comparisons, the development performed for the previous transient was kept, although it was not necessary. The additional run-back did not produce either a low or high SG level. Cases (a) and (b) were so similar that we do not show a figure for the latter, although a certain improvement of the margin was appreciated. Calculation (b) provided the answer to question (4).

In calculation (c) both manual actions were simulated as they occurred in the actual transient:

(i) additional manual run-back at 71 seconds,
(ii) FW 3 partial manual closure at 150 seconds.

The results can be appreciated in Table 2 and Figures 4–7.

The observed inflexion C (Figure 7) finally was the result of two combined effects. C was partially due to the impact of the automatic closure of loops 1 and 2 on loop 3. Inflexion C was also caused by the fine mechanism of partially closing a valve from the control room. This operation always shows a period of closure and a subsequent release of the driving device. Different attempts had been tested reasonably combining different times of closure/release. Calculation (c) uses the best combination of them and explains what could have happened in the plant.

Therefore, in order to draw conclusions on the actual capabilities of a protection system, many apparently nonsignificant engineering features must be clarified. The information produced allows the adequacy of run-back design to be corroborated and helps to enhance the database of useful experiences.

The follow-up actions arising from this analysis started from the fact that design adequacy is confirmed by the results of the calculations. No actions were taken on cause analysis, or design limits. Furthermore, no design or procedure modifications were needed. The only action taken was focused on informing on "lessons learned" about run-back effectiveness and on the fine behavior of main FW valves.

The transient, in any case, was significant enough to be included in the qualification matrix of the Ascó NPP plant model.

5. EXAMPLE OF A HYPOTHETICAL TRANSIENT ANALYSIS: LOSS OF MAIN FW WITHOUT SCRAM

The scenario selected for this example was related to the characterization of ATWS mitigation system actuation circuitry (AMSAC), a system designed to mitigate the consequences of anticipated transients without scram (ATWS). It is an example of a calculation [9] aimed at providing the grounds for dialogue with the safety authority. AMSAC adequacy is regulated following a procedure based on two different premises

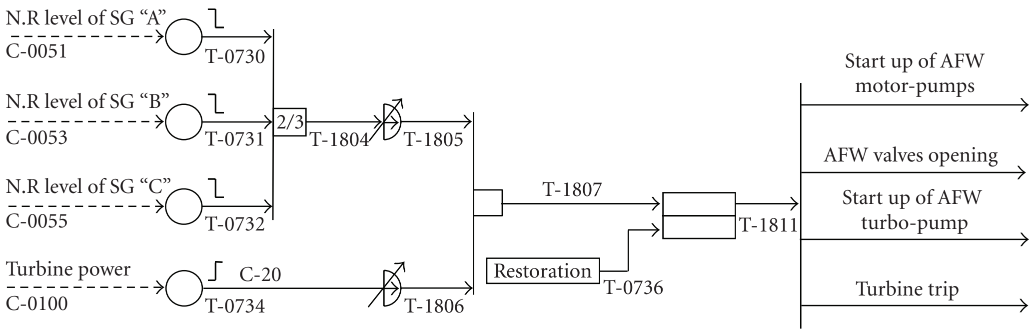

Figure 8: Logic diagram of the AMSAC system.

Table 2: Sequence of events for FW turbo-pump actual transient.

Event	Recorded time (s)	Calculated time (s)
Main FW turbo-pump stop	21	21
Run-back automatic signal	22	22
Additional manual run-back	71	71
Manual closure of FW 3	150	150
Automatic closure signal of FW1 and 2	201	196
Reactor scram due to high SG3 level	240	249

or statements considered as the starting point of the analysis/discussion. The premises are

(i) an existing generic assessment establishes the effectiveness of the combined effect of the AMSAC and inherent nuclear feedback [20] such as moderator temperature coefficients of current core designs to reduce power thus ensuring that primary pressure peak remains below design limits,

(ii) PSA models show reasonably high results for reactor protection system (RPS) reliability.

These two statements are accepted and supported by previous calculations that are not part of this analysis. The first statement defines the generic licensing and the second one is a kind of requirement needed to start dialogue.

The current analysis is devoted to extend generic results to future core designs. Thus, if future core designs maintain or improve feed back effects of current design, the effectiveness of mitigating actions is ensured. All of this will induce the regulatory body and licensee to discuss how a plant such as Ascó or Vandellòs-II (with a definite core design and relief capacity) fits within the generic assessment.

Transients without scram are managed by using core feedback effects. AMSAC protection is designed to improve the success of the strategy. In any transient in which primary pressure and temperature are allowed to increase, there is a risk of overpressure that needs to be both studied and controlled.

The general philosophy of the protection is to use the effects of fuel and moderator feedback in order to produce a power decrease at an initial stage, and subsequently, at a second stage, to allow normal relief systems to maintain primary pressure within mechanical limits established by the ASME code. Thus, protection features together with relief capacity have been tested in this study.

The transient presented below is a loss of FW without scram, as this case is traditionally considered to be the most crucial scenario among those initiated by a condition II event. For a given plant, it must be demonstrated that by following this event and assuming RPS failure, the reactor power decreases and primary pressure does not exceed the ASME limit.

The logic of the protection system can be followed in the corresponding diagram (Figure 8). An AMSAC signal is produced when 2 out of 3 narrow-range SG level signals become lower than the so-called low-low set point. This signal is delayed a few seconds and it activates the turbine trip and the automatic start-up of the auxiliary feed-water (AFW). At a first stage the turbine trip produces a pressure and temperature increase leading to power reduction (through feed-back effects) and the AFW start-up at the second stage helps to recover the plant. The delay must be properly tuned in order to fulfil the mitigation goal.

In order to simulate the transient and to characterize the protection actuation, some relevant aspects have to be taken into account, basically neutronic feed-back, signal delay, primary relief capacity, heat transfer from primary to secondary side, and secondary relief capacity.

Neutron-kinetics and thermal-hydraulics are coupled in this case. Fortunately, the transient is symmetric and zero-D kinetics with enhanced fuel and moderator tables were the suitable option. Although standard information from the core designer [21] is related to operation ranges for reactivity feedback, improved tables were available covering wide ranges of moderator densities. Fuel effect is also relevant, but it was not necessary to include additional information to the kinetic model.

Several calculations were performed in order justify the signal delay proposed by the designer. Once the behavior of the involved systems was characterized, two calculations were performed: one with and one without protection actuation. For simplicity, only the results of the first calculation are presented. Figure 9 shows moderator average temperature and nuclear power. An increase in the moderator

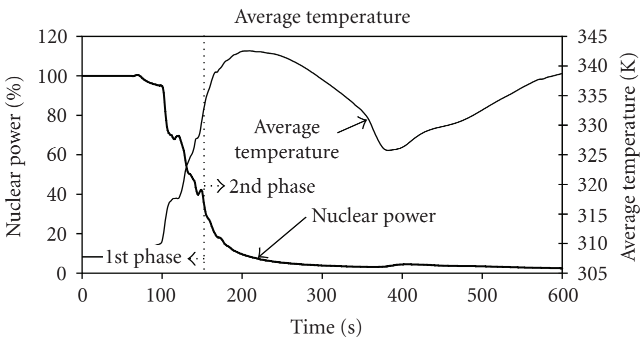

Figure 9: Average temperature and nuclear power (AMSAC transient).

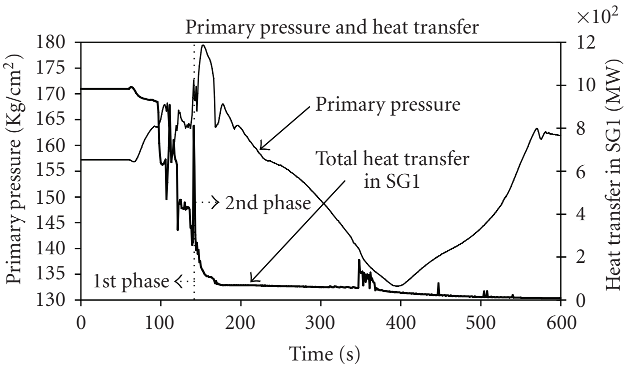

Figure 10: Heat transfer in SG1 and primary pressure (AMSAC transient).

temperature results in a decrease of nuclear power. Figure 10 shows heat transfer in SG1 and primary pressure. Heat transfer decrease during the first phase causes an increase in average moderator temperature. Heat transfer in the second phase remains at a nonzero value, which contributes to energy extraction and results in a primary pressure peak of about 17.95 MPa, which is clearly lower than the ASME limit (3200 psia or 22.06 MPa). The pressure peak evaluated in the base calculation without AMSAC activation showed a value of 21.81 MPa.

The results prove, for the considered core design,

(i) the effectiveness of AMSAC system,
(ii) the capability of relief and safety valves to mitigate pressure peak.

The study is an illustration of the usefulness of a BE plant model with a rather simple but effective coupled neutronics. It is also an example of how calculations result could provide the basis for dialogue with the regulator. The BE prediction of the scenario helps characterizing the interactive behavior of the involved systems.

6. EXAMPLE OF AN EOP/PSA TRANSIENT ANALYSIS: TOTAL LOSS OF FW

EOP/PSA transient analyses are the most usual studies that are traditionally performed using integral plant models.

Analytical support for EOPs development is a very complex task requiring a great deal of effort. A specific IAEA report [23] establishes the tasks related to such activity which currently involves different organizations. ANAV-UPC coordinated team has been in many occasions the responsible of validation analyses.

The selected group of scenarios for this example is the "total loss of FW," which occurs due to a main FW turbopump trip or due to a malfunction of the main FW valves [10].

The generic information available consists of the description of the feed and bleed (F&B) procedure, as well as the description of the systems involved.

The simultaneous failure of the AFW system causes loss of the heat sink and, shortly after, both the turbine and the reactor trip.

The operators start following the EOP "*Reactor trip and/or Safety Injection*." The first steps of this EOP verify the function of the AFW and try to ensure recovery. As Ascó EOPs are symptom oriented, the minimum time for transfer to the specific EOP "*loss of heat sink*" is quite long (about 10 minutes). For all this period of time, the level of the steam generators will uniformly decrease. Once the wide-range level of 2 out of 3 SGs becomes less than 6%, it is time to start the F&B procedure: reactor coolant pumps (RCPs) are tripped, 2 out of 2 pressurizer power operated relief valves (PORVs) are opened and high pressure injection system (HPIS) is activated.

After a period of time, the plant will be cooled down and the final steps of the procedure aim to properly stop the HPIS and to close PORV once the plant has been recovered (EOP "*Finalizing Safety Injection*").

The objective of the analysis is to prove the effectiveness of the primary F&B procedure and to answer questions and subjects, set by both the operation and PSA groups, related to

(i) timing evaluation,
(ii) possibility of successfully executing the procedure having only 1 relief valve available.

The scenario selected as the *base case* for this analysis has the following features and assumptions:

(i) loss of FW at time 50 seconds,
(ii) failure of 1 out of 2 HPIS trains and the availability of only one single PORV,
(iii) AFW and steam-dump unavailable,
(iv) no recovery actions are assumed.

Manual actions are the following:

(i) at time 350 seconds RCPs are stopped and their coast down is initiated;
(ii) time to start the procedure is set to the minimum reasonable time, 600 seconds, (i.e., 600 seconds after the wide-range level of 2 out of 3 SGs reaches 6%, PORV is opened and HPIS is actuated).

After the total loss of FW takes place, heat transfer from the primary to the secondary side degrades and causes a decrease of the SG level. Once this symptom has been detected,

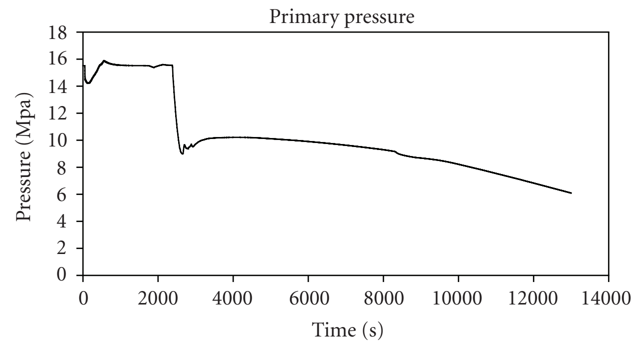

Figure 11: Primary pressure (F&B base case).

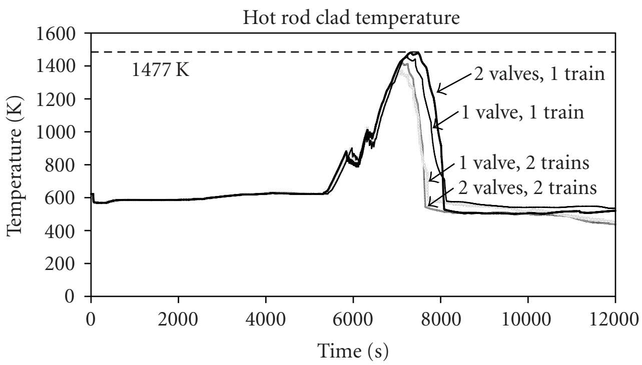

Figure 12: Hot rod clad temperature (F&B availability study).

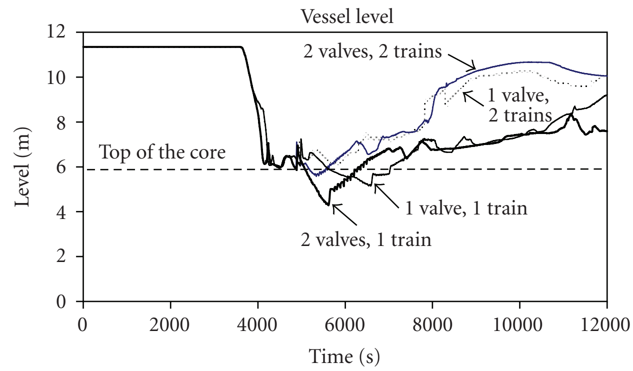

Figure 13: Reactor vessel level (F&B availability study).

the procedure starts by opening 1 PORV and actuating 1 HPIS train. Water injected into the primary system at low temperature is heated by decay power and comes out through the relief valve. The procedure results in a pressure decrease (see Figure 11), which means that energy produced is completely extracted.

The base case brings the plant to a safe situation without violating design limits as hot rod clad temperatures show a general decreasing trend during the whole transient. The calculation properly captures the main relevant thermal-hydraulic features of the scenario.

Once the base case has been successfully simulated, a *strategy* is defined to answer the following:

(i) impact of PORV and HPIS partial availability (less than 2 PORV or 2 HPIS trains),

(ii) maximum time to start the procedure after the level symptom occurs,

(iii) relevant heat sink recovery phenomena (although recovery actions are quite fast, they involve different components and need some time).

If the answers to the questions above are obtained, the operation team will have a better general picture of the scenario and related phenomena. As obviously each answer has an impact on the others, the strategy applied is to launch quite a large number of combined scenarios in order to cover different situations that could potentially occur. For a given combination of component availability, a series of different procedure starting times have been tried and for each of these calculations heat sink recovery was also imposed at different times.

The total number of cases was 61. The following paragraphs show the most relevant results of the study.

Two findings were identified in the sensitivity study regarding the impact of the partial availability of PORVs and HPIS trains. Stress must be laid on the word “identified” as this is the aim of a study oriented, as a first approach, to operation support.

(i) The first finding is related to the connection between operation and PSA. With the availability of one PORV and one HPIS train, the plant can be recovered (Figure 12). Although this statement is made based on multiple calculations sweeping over different ranges of operation boundary conditions, it is still pending further analysis (basically uncertainty evaluation).

(ii) The second is related to an interesting phenomenon that takes place in the transients with the availability of 2 valves and 1 train. In this transient the depressurization rate is high (2 valves) and water supply is low (1 train). High depressurization instantly affects all the primary circuit, produces a lower saturation temperature and helps steam generation in the core. This result does not seem critical at all, as it only causes a small peak of temperature quite within design limits and it is a useful result for operation, as it explains a non-intuitive situation: the Reactor Vessel Level Instrumentation System (RVLIS) implemented in Ascó NPP can supply a lower level value in a situation with higher availability of components (see Figure 13).

A second sensitivity study was performed to establish the maximum time to start the procedure.

Figure 14 shows the hot rod clad temperature for a selected group of transients described in Table 3. In 2 out of 4 cases (time to start procedure after symptom equals 600 s and 3000 s) the maximum temperature remains below 1477°K (2200°F). In the case of a delay of 5700 s the clad temperature goes slightly beyond this limit and plant recovery is not successful. In between, there is a case that needs further analysis (a delay of 5100 s).

Table 3: Features of selected transients (F&B starting time analysis).

	Starting time of the F&B procedure (s)			
	Case 1 600 s	Case 2 3000 s	Case 3 5100 s	Case 4 5700 s
Main FW turbo-pumps stop	50	50	50	50
Turbine and reactor stop	50	50	50	50
Manual action: RCP stop	350	350	350	350
2 out of 3 SG wide range level reach 6%	1782	1782	1782	1782
Manual action: PORV is opened	2382	4782	6882	7482
Manual action: HPIS is actuated	2382	4782	6882	7482
Clad maximum temperature is reached (clad maximum temperature)	48 (622 K)	4220 (628 K)	7240 (1460 K)	7610 (1652 K)
Transient end	13000	13000	13000	13000

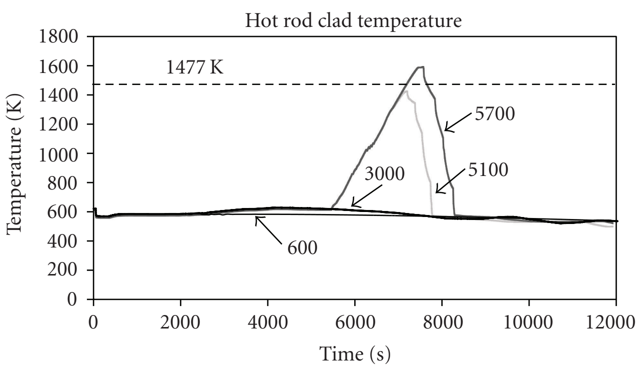

Figure 14: Hot rod clad temperature (F&B starting time analysis).

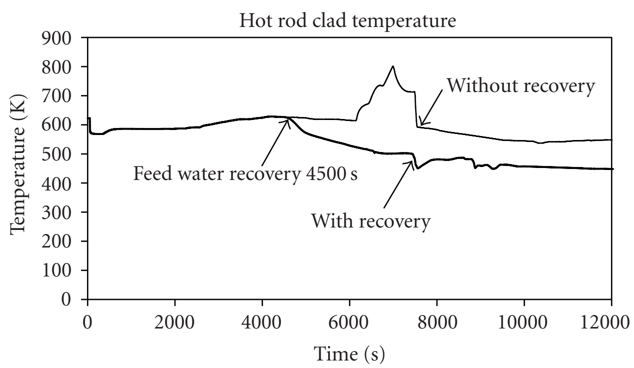

Figure 15: Hot rod clad temperature (F&B heat sink recovery analysis).

This study establishes a first approach for the maximum time available for starting the F&B procedure (5100 s) and bringing the plant to a safe situation. This result is interesting not only for PSA, but also for operation and training. Nevertheless, as it is just a first-approach calculation, it needs to be confirmed after evaluating the associated uncertainty.

The sensitivity study aimed at analysing heat sink recovery actions provides quite useful information for operators, as it establishes the relationship between the actions performed and trends and data that can be observed in the control room. This point is especially interesting in operation and training, even though Ascó EOPs are symptom oriented.

The clad temperatures of the selected case with and without heat sink recovery are shown in Figure 15. On the one hand, the transient without heat sink recovery shows a short partial core uncover with a corresponding temperature increase, but ends in a safe situation. On the other hand, the transient with heat sink recovery at 4500 s completely prevents the mentioned increase.

The results obtained in the study of the group of scenarios of "Total loss of feed water" for Ascó NPP are valuable for the safe operation of the plant.

The analysis provides answers for different operation questions about the studied scenarios and produces a better general picture of the group of transients and related phenomena. It also helps to get a better understanding of PSA results, as they are corroborated.

The analysis identifies, among all the transient runs performed, those that require further study of the uncertainty evaluation. Future work on this point can be straightforwardly directed to recognized calculations.

7. CONCLUSIONS

Dynamic analysis supporting plant operation is an engineering task that shares objectives with other engineering branches that support plant operation. It is connected with different technical features of plant design and produces results that are useful for safety, operation, design and training.

ANAV has had an important advantage by having its own analysts on its technical staff. As members of the corresponding engineering team, the analysts have been essential in order to smooth the relationship between different organizations involved in the most important decisions taken related to the operation and safety of the Ascó and Vandellòs-II reactors.

The main tool of this type of analysis has been the integral plant model. After 15 years of these practices, nodalizations have been maintained and improved at a quality level to ensure optimum performance. During all this time ANAV analysts have worked together with the UPC team at different levels. Innovative engineering and research compose the scope of analytical activities that have resulted fruitful for both the utility and the university.

Anticipating expected behaviour has revealed itself to be extremely useful for operation support and decision-making. Some results of the analysis performed were crucial at the time they were produced, such as the impact of SG tube plugging on the dynamic behaviour of Ascó NPP. Some were complete and helpful for safety, such as support to the licensing of the AMSAC system. Some others were a combined effort by different organizations, such as the analytical support to SG substitution at Ascó NPP. The more the nodalizations are used by qualified users with a deep knowledge of them, the more accurate and useful they become, not only for safety issues, but also for issues related to operation and engineering.

The study fulfils its objective of illustrating the usefulness of computational analysis for operational support.

ACRONYMS

AFW:	Auxiliary feed water
AMSAC:	ATWS mitigation system actuation circuitry
ANAV:	Asociación Nuclear Ascó-Vandellòs
ATWS:	Anticipated transient without scram
ASME:	American standards of mechanical engineering
BE:	Best estimate
BEPU:	Best estimate plus uncertainty
EOP:	Emergency operating procedures
F&B:	Feed and bleed
FW:	Feed water
HPIS:	High pressure injection system
NPP:	Nuclear power plant
PORV:	Power operated relief valves
PSA:	Probabilistic safety analysis
PWR:	Pressurized water reactor
RCP:	Reactor coolant pump
RPS:	Reactor protection system
RVLIS:	Reactor vessel level instrumentation system
SG:	Steam generator
UPC:	Technical University of Catalonia

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Review Article

An Overview of Westinghouse Realistic Large Break LOCA Evaluation Model

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Since the 1988 amendment of the 10 CFR 50.46 rule in 1988, Westinghouse has been developing and applying realistic or best-estimate methods to perform LOCA safety analyses. A realistic analysis requires the execution of various realistic LOCA transient simulations where the effect of both model and input uncertainties are ranged and propagated throughout the transients. The outcome is typically a range of results with associated probabilities. The thermal/hydraulic code is the engine of the methodology but a procedure is developed to assess the code and determine its biases and uncertainties. In addition, inputs to the simulation are also affected by uncertainty and these uncertainties are incorporated into the process. Several approaches have been proposed and applied in the industry in the framework of best-estimate methods. Most of the implementations, including Westinghouse, follow the Code Scaling, Applicability and Uncertainty (CSAU) methodology. Westinghouse methodology is based on the use of the WCOBRA/TRAC thermal-hydraulic code. The paper starts with an overview of the regulations and its interpretation in the context of realistic analysis. The CSAU roadmap is reviewed in the context of its implementation in the Westinghouse evaluation model. An overview of the code (WCOBRA/TRAC) and methodology is provided. Finally, the recent evolution to nonparametric statistics in the current edition of the W methodology is discussed. Sample results of a typical large break LOCA analysis for a PWR are provided.

1. INTRODUCTION

The 1988 amendment of the 10 CFR 50.46 rule allowed the use of realistic physical models to analyze loss-of-coolant accident (LOCA). Best-estimate LOCA methods are now extensively employed within the nuclear industry. In particular, Westinghouse has been developing and applying realistic or best-estimate LOCA methods for almost two decades now and a large amount of experience has being gained in this field.

The Westinghouse realistic (best-estimate) methodology is based on the Code Scaling, Applicability and Uncertainty (CSAU) methodology (Boyack et al. [1]). The methodology was approved by the NRC in 1996 after an extensive review. At that time, this was the first best-estimate (BE) LOCA evaluation model approved (Bajorek et al. [2], Young et al. [3]). In its original version W BE methodology was applicable to 3- and 4-loop plants with safety injection into the cold leg. Subsequently, the methodology applicability was extended to 2-loop plants with upper plenum injection (UPI) in 1999 (Takeuchi et al. [4–6]) and advanced passive plant such as the AP600 and AP1000 (Frepoli et al. [7]). Since its approval, Westinghouse has applied the methodology to more than 30 nuclear power plants (Muftuoglu et al. [8], Frepoli et al. [9–11]) both in the USA and abroad.

Westinghouse LOCA methodology is based on the use of WCOBRA/TRAC computer code. Sections 3 and 4 provide an overview of code features, its assessment basis, and identified source of biases and uncertainties.

A key step in a best-estimate analysis is the assessment of uncertainties associated with physical models, data uncertainties, and plant initial and boundary condition variabilities. As uncertainties are incorporated into the process, a procedure is developed where the results from several calculations are collected to develop a statement where compliance with prescriptive rules or acceptance criteria is demonstrated. Based on the current 10 CFR 50.46 rule, an emergency core cooling system (ECCS) design is required to satisfy three main criteria: (1) the peak clad temperature (PCT) should be less than 2200 F, (2) the local maximum clad oxidation (LMO) should be less than 17%, and (3) the core-wide oxidation (CWO) should be less than 1%. More insights on the regulations and how industry satisfies those rules in the framework of realistic calculations are provided in Section 2.

The technique used to combine those uncertainties evolved over the years. In its original implementation, Westinghouse methodology followed strictly CASU where the use of response surface was suggested as a practical means to combine the various uncertainty components. More recently, the methodology was modified toward nonparametric methods. The current methodology is called Automated Statistical Treatment of Uncertainty Method (ASTRUM) (Nissley, et al. [12], Frepoli and Oriani [13]). The main difference between the new and the old techniques is in the evaluation of final uncertainty, Element III of CSAU. A comparison between the two techniques is discussed by Muftuoglu et al. [8]. A review of these techniques is given in Section 5 while sample results are provided in Section 6.

2. HISTORICAL BACKGROUND AND REVIEW OF REGULATIONS

A large-break-LOCA event is categorized as a design-basis accident. The current safety regulations of the United States Nuclear Regulatory Commission (US NRC) are stipulated in 10 CFR Part 50, Section 50.46. Based on the 10 CFR 50.46 rule, an emergency core cooling system (ECCS) design is required to satisfy prescriptive criteria. The regulation identifies the following five criteria.

(1) Peak clad temperature (PCT) should be less than 2200 F.
(2) Local maximum oxidation (LMO) should be less than 17%.
(3) Core-wide oxidation (CWO) should be less than 1% (to limit the maximum amount of hydrogen generated).
(4) The core should maintain a coolable geometry.
(5) Long-term cooling should be demonstrated.

Typically, the last two criteria (coolable geometry and long-term cooling) are satisfied outside the LOCA analysis once the LOCA calculation demonstrate to be in compliance with the first three criteria.

The acceptance criteria above were established following an extensive rulemaking in 1973. Also the regulation at that time was formulated to account of potentially unknown phenomena and recognizing lack of knowledge of fundamental physical phenomena. Several conservative "required features" were mandated in Appendix K to 10 CFR 50. To cite some, the decay heat was based on ANS 1971 model + 20%; the metal-water reaction calculation was based on the conservative Baker-Just model; the heat transfer was limited to steam only for low-flooding rates; and so on.

This led to broad international development efforts to better understand LOCA phenomena and processes, in particular the large break LOCA. The effort was both on the experimental side and analytical side (computer codes, evaluation models). The major contributor to the development effort was the international 2D-3D program which focus on multidimensional phenomena and scaling considerations. The test facilities are full-scale upper plenum test facility (UPTF); large-scale cylindrical core test facility (CCTF); slab core test facility (SCTF).

The knowledge gained over the years led the industry to consider a more realistic approach in the analysis of the LOCA scenario (ECCS [14]). In 1988, the USNRC amended its regulations (10 CFR 50.46) to allow the use of realistic physical models (Federal Register [15]), simulated in computer codes, to analyze the loss-of-coolant accident (LOCA) in a PWR. In the amended rule, the acceptance criteria were not changed (PCT = 2200 F, LMO = 17%, and CWO = 1%), however certain physical models were identified as acceptable but not prescribed. Acceptable data sources were identified and documentation requirements specified (Regulatory Guide 1.157). Any realistic calculation requires the assessment of the uncertainties. Overall requirements for quantifying uncertainties were specified and the Code Scaling, Applicability and Uncertainty (CSAU) method (Boyack et al. [1]) was cited CSAU as acceptable methodology framework. An overview of the CSAU process is given in the next section.

2.1. *Overview of the code scaling, applicability and uncertainty (CSAU) roadmap*

A group of experts (referred to as the technical program group or TPG) under the sponsorship of the US Nuclear Regulatory Commission (USNRC) took an effort to demonstrate that practical methods could be developed which would be acceptable under the new regulations. Shortly after its completion, the CSAU methodology and its demonstration were described in a series of papers appearing in Nuclear Engineering and Design (Boyack et al. [16, 17]).

The CSAU process is divided in three main elements. In Element (1), the scenario is broken down into relevant-time periods (e.g., blowdown, refill, and reflood for large-break scenario) and the nuclear power plant broken down into relevant regions (e.g., fuel rod, core, lower plenum). Then potentially important phenomena/processes are identified for each time period and region. An expert's panel performs ranking and document basis for consensus. Results are compiled in the phenomena identification and ranking table (PIRT). The PIRT is a critical element of CSAU-based methodologies. It is designed to focus the prioritization of code assessment and facilitate the decisions on physical model and methodology development.

Element (2) is the assessment of the code. An assessment matrix is established where separate effect tests (SETs) and integral effect tests (IETs) are selected to validate the code against the important phenomena identified in the PIRT. The code biases and uncertainties are established and the effect of scale determined. A key output from this element is the establishment of probability distributions and biases for the contributors identified in Element (1). In addition to the generation of probability distributions, and perhaps even more important, this element required a thorough assessment of the code's ability to correctly predict all the dominant physical processes during the transient. This leads to the adequacy decision of the evaluation model.

Element (3) is the actual implementation stage of the methodology. Sensitivity and uncertainty analyses are performed here. This element is probably the most straight forward of all the elements. The dominant contributors and their probability distributions are properly identified and quantified, and if the computer code, through assessment and comparison with data, is shown to accurately predict the effect of variations in input variables on the output result, then several well-established methods are available to perform the uncertainty propagation step. The choice of method is basically a practical one, controlled by the expense incurred in performing computer calculations. The methods utilized evolved over the last two decades. An overview of the methods for combining the uncertainties is provided in Section 5.

The CSAU is a practical roadmap to develop a realistic methodology but shortcomings were recognized since its introduction. In particular, with regard to the PIRT, the human judgment factor and the fact that knowledge gained is not always factored back into final documentation were seen as a point of weakness. Soon after its introduction, the CSAU methodology was reviewed by the technical community, and comments were published in Nuclear Engineering and Design (Hochreiter [18]). Although there was agreement that the methodology described many of the key steps required for an acceptable methodology, there was also technical criticism and some skepticism on the practical applicability of the methodology (Boyack et al. [17]).

One important issue raised was whether the PIRT procedure eliminated too many important processes from consideration. This concern is heightened by the fact that since every additional process which is included increases the complexity and cost of subsequent steps, there is the possibility of 'rationalizing' a short list of contributors.

However, there are three conditions preventing such an occurrence: First, detailed independent review of the methodology by the USNRC's experts eventually brings to light important processes which may have initially been ignored. Second, [19] provides a complete list of all the processes known to affect the LOCA transient, and requires a detailed assessment of each one. Third, the CSAU methodology requires thorough assessment of a "frozen" version of the computer code with a wide variety of experiments. Since these experiments are specifically selected to cover the expected range of conditions, important phenomena will be identified.

Overall, an important claim made by the TPG was that the methodology was structured, traceable, and practical and therefore it was ideally suited for application in the regulatory and design arenas. This was definitely demonstrated by several successful implementations of the CSAU-based methodologies currently licensed and applied to safety analysis in the industry.

Beginning in the mid 1980s, Westinghouse began development of a best-estimate methodology, in partnership with the Electric Power Research Institute and Consolidated Edison (Calif, USA). Acceptance of the methodology was achieved in 1996 after a rigorous review spanning over 3 years. A summary of the technical review and the conditions of acceptance was issued by the USNRC (Jones and Liparulo [20]). Many of the questions raised by the technical community concerning the CSAU methodology were dealt with during this review.

The PIRT concept has evolved over years (Wilson et al. [21] and Boyack et al. [22]) and has been extensively used in various areas by the industry. Main area of application is the development of realistic analysis methodologies (not limited to LOCA) and the development of testing requirements for new plant designs. Recent PIRT also includes the "state of knowledge." This process puts significant emphasis on processes or phenomena that are flagged as highly important with a low state of knowledge.

The CSAU was recently endorsed as an acceptable structured process in the recently published Standard Review Plan (NUREG-0800) [23] and Regulatory Guide 1.203 (2005) [24]. In particular, RG 1.203 describes a structured evaluation model development and assessment process (EMDAP) which essentially follows the same principles of the CSAU roadmap with more emphasis given to the evaluation model development process which starts from the definition of the objectives, the functional requirements, and the assessment and leads to the evaluation model adequacy decision. The EMDAP process is depicted in the flowchart of Figure 1.

2.2. Regulations within a statistical framework

While Elements (1) and (2) of the CSAU are generally applied in various form consistently with the original intent, the techniques used to combine the uncertainties evolved over the last few years. The CSAU originally suggested the use of response surfaces methods, however shortcomings were soon identified in early implementation. Direction in recent years is toward direct Monte Carlo methods and the use of nonparametric statistics. This generated a debate in the industry since the regulations are not directly suited to a statistical framework. A discussion on the interpretation of the regulations from this perspective is presented in this section.

The key step in a realistic analysis is the assessment of uncertainties associated with physical models, data uncertainties, and plant initial and boundary condition variabilities. The issue is how results are interpreted to demonstrate compliance with the 10 CFR 50.46 requirements. As an additional requirement/clarification, 10 CFR 50.46 states that "*[· · ·] uncertainty must be accounted for, so that, when the calculated ECCS cooling performance is compared to the criteria set forth in paragraph (b) of this section, there is a high level of probability that the criteria would not be exceeded.*" Paragraph (b) of 10 CFR 50.46 contains the list of the acceptance criteria. 10 CFR 50.46 does not explicitly specify how this probability should be evaluated or what its value should be.

Additional clarification as to the US NRC expectations on the acceptable implementation of the "high probability" requirement is provided in Section 4 of Regulatory Guide 1.157 (Best-estimate Calculations of Emergency Core Cooling System Performance) that states "*a 95% probability is considered acceptable by the NRC staff [· · ·].*"

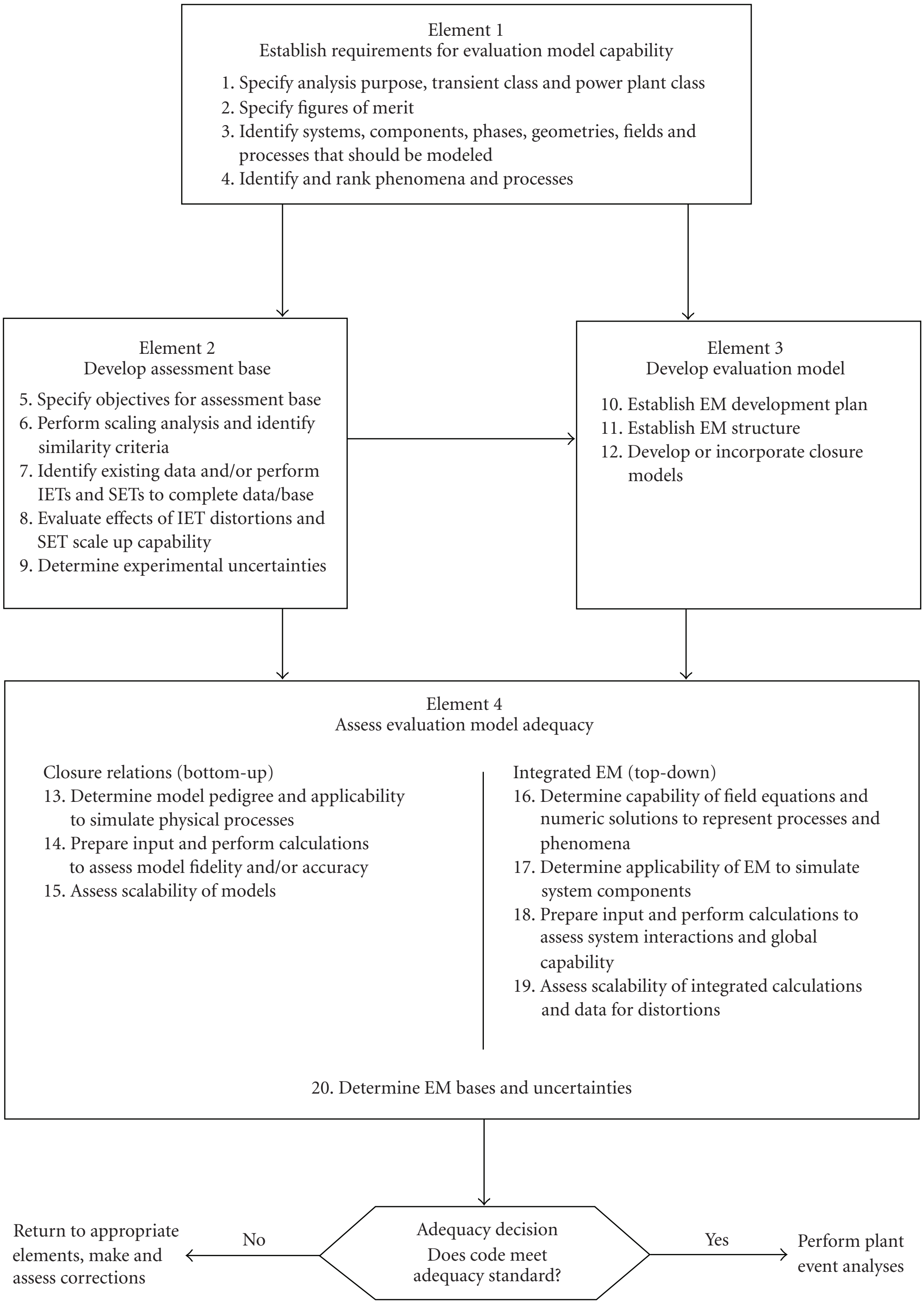

Figure 1: EMDAP (Reg. Guide 1.203).

The regulatory guide was not developed to the point of explicitly considering a statistical approach to the uncertainties treatment, which would also require a statement with regard to the confidence level associated with a statistical estimate of the uncertainty. Regulatory Guide 1.157 introduced the concept of confidence level as a possible refinement to the uncertainty treatment, but did not expand further on this concept.

As statistical methods are implemented to perform LOCA safety analyses, a statistical statement based on a 95% confidence level has been suggested by the NRC as acceptable. This will be discussed further in Section 5. In practice, a 95% confidence that the 95th percentile of PCT, LMO, and CWO populations is within the specified acceptance criteria is considered acceptable by the USNRC to demonstrate the required "high probability." In particular the safety evaluation report (SER) of the Westinghouse best-estimate large break LOCA methodology (ASTRUM) states the following: "*the staff determined that a 95th percentile probability level based on best approximations of the constituent parameter distributions and the statistical approach used in the methodology is appropriately high.*"

The main reason that a 95/95 statistical statement is accepted lies in the defense-in-depth philosophy. It is recognized that many other layers of conservatisim are included in any licensed realistic evaluation model. For example, the following is stated by the NRC in ASTRUM SER: "*Because this application only applies to LBLOCA design basis analyses (which assume a single failure), a higher probability [· · ·] is not needed to assure a safe design.*" Note that the single failure assumption is not the only conservative bias/assumption included in the Westinghouse methodology. The use of this and other conservative assumptions further supports the conclusions that a 95/95 statistical statement is adequate to satisfy the acceptance criteria, for the proposed evaluation model.

3. THE ENGINE OF W METHODOLOGY: WCOBRA/TRAC COMPUTER CODE

Westinghouse large break LOCA evaluation model is based on the use of the WCOBRA/TRAC thermal-hydraulic code, the engine of the methodology. This code was developed from COBRA/TRAC which was originally developed at Pacific Northwest Laboratory (Thurgood et al. [25]) by combining the COBRA-TF code (Thurgood et al. [26]) and the TRAC-PD2 codes (Liles et al. [27]). The COBRA-TF code, which has the capability to model three-dimensional flow behavior in a reactor vessel, was incorporated into TRAC-PD2 to replace its vessel model. TRAC-PD2 is a system-transient code designed to model all major components in the primary system. Westinghouse continued the development and validation of COBRA/TRAC through an extensive assessment against several separate effect tests (SETs) (Paik and Hochreiter [28]) and integral effect tests (IETs).

The COBRA-TF (3D Module) is based on a two-fluid, three-field representation of two-phase flow. The three fields are a vapor field, a continuous liquid field, and an entrained liquid drop field. Each field in the vessel uses a set of three-dimensional continuity, momentum, and energy equations with one exception: common energy equation is used by both the continuous liquid and the entrained liquid drop fields. The one-dimensional components (TRAC-PD2) consist of all the major components in the primary system, such as pipes, pumps, valves, steam generators, and the pressurizer. The one-dimensional components are represented by a two-phase, five-equation, drift flux model.

Among the new models and improvements incorporated by Westinghouse are (1) improved DFFB (dispersed flow film boiling); (2) bottom/top downflooding (Reflood Entrainment); accumulator nitrogen model; (3) a new core kinetic model (point kinetic); (4) spacer grid model which includes the heat transfer enhancement, drop breakup and grid rewet effects; (5) a two-fluid choke flow model based on TRAC-PF1 formulation (Liles et al. [29]); (6) an improved fuel rod model; (7) upgraded interfacial drag models.

The subchannel formulation included in the 3D module (COBRA) offers a large flexibility from the modeling stand point (Figure 2). The geometric complexity of the vessel internals and hardware can be modeled with great details with a relative coarse hydraulic mesh. For example, important is the capability of explicitly modeling the hot assembly within the core.

Westinghouse followed the PIRT process to identify and rank dominant phenomena. Important phenomena identified were as follows.

(1) Break flow.
(2) Break path resistance.
(3) Initial stored energy/fuel rod.
(4) Core heat transfer.
(5) Delivery and bypass of ECCS water.
(6) Steam binding/entrainment.
(7) Condensation in cold leg and downcomer.
(8) Noncondensable gases/accumulator nitrogen effects.

Note that several additional contributors not considered important in the CSAU demonstration were identified by Westinghouse. Two examples are the effect of the broken loop resistances such as the pump and vessel nozzles on the core-ow rate, and the effect of fuel relocation after cladding burst on local linear power. It was also found that it is important to consider the effect of variations in plant-operating conditions such as the core power distribution and transient peaking factors allowed by the technical speci cations for the plant. In the CSAU demonstration, this aspect was not given much attention.

For large break LOCA application, more than 100 tests and 20 facilities were simulated by WCOBRA/TRAC. Quantifications of model uncertainty such as heat transfer and critical flow were performed via SETs. IETs and large component tests were used for judging the code's ability to predict system responses. This includes the effect of the noding. In particular, PWR noding is consistent with noding used in code assessment as much as in the practical part. Compensating error analyses were performed to investigate the interaction of various models and identify situations where an apparently good prediction is due to offsetting mispredictions. Figure 3 shows the typical WCOBRA/TRAC vessel noding.

The influence of the user on the results has been recognized as another potential source of uncertainty (Aksan et al. [30], Glaeser [31–33]). To eliminate such variability, several engineering safeguards or procedures are considered as part of the methodology. Calculation of plant-specific inputs and setup of initial and boundary conditions follow a very prescriptive standard guidance which is formulated in standard procedures. Frequent engineering and peer reviews are implemented to assure adherence to this guidance. In this framework, plant-to-plant variations are limited as much as in the practical part. Steady-state criteria are established to minimize variability of initial conditions. Following this

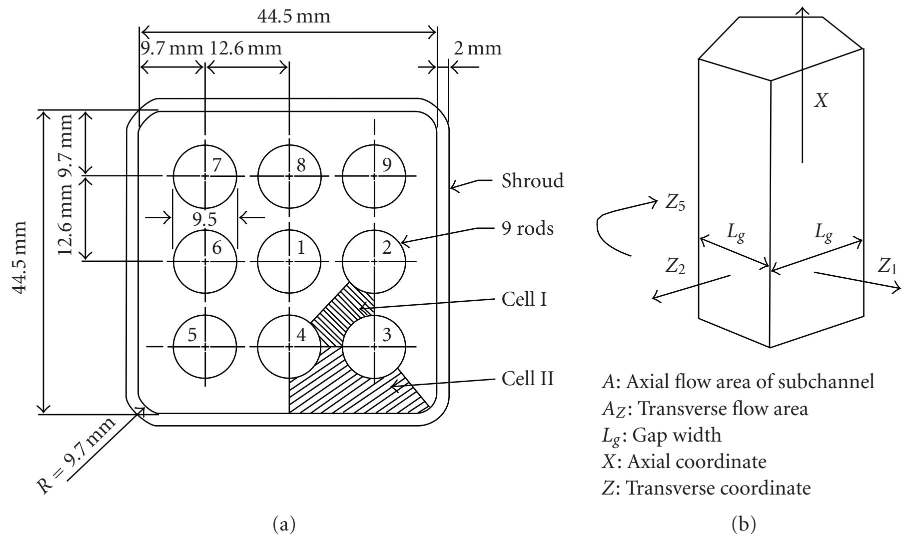

Figure 2: WCOBRA/TRAC subchannel formulation.

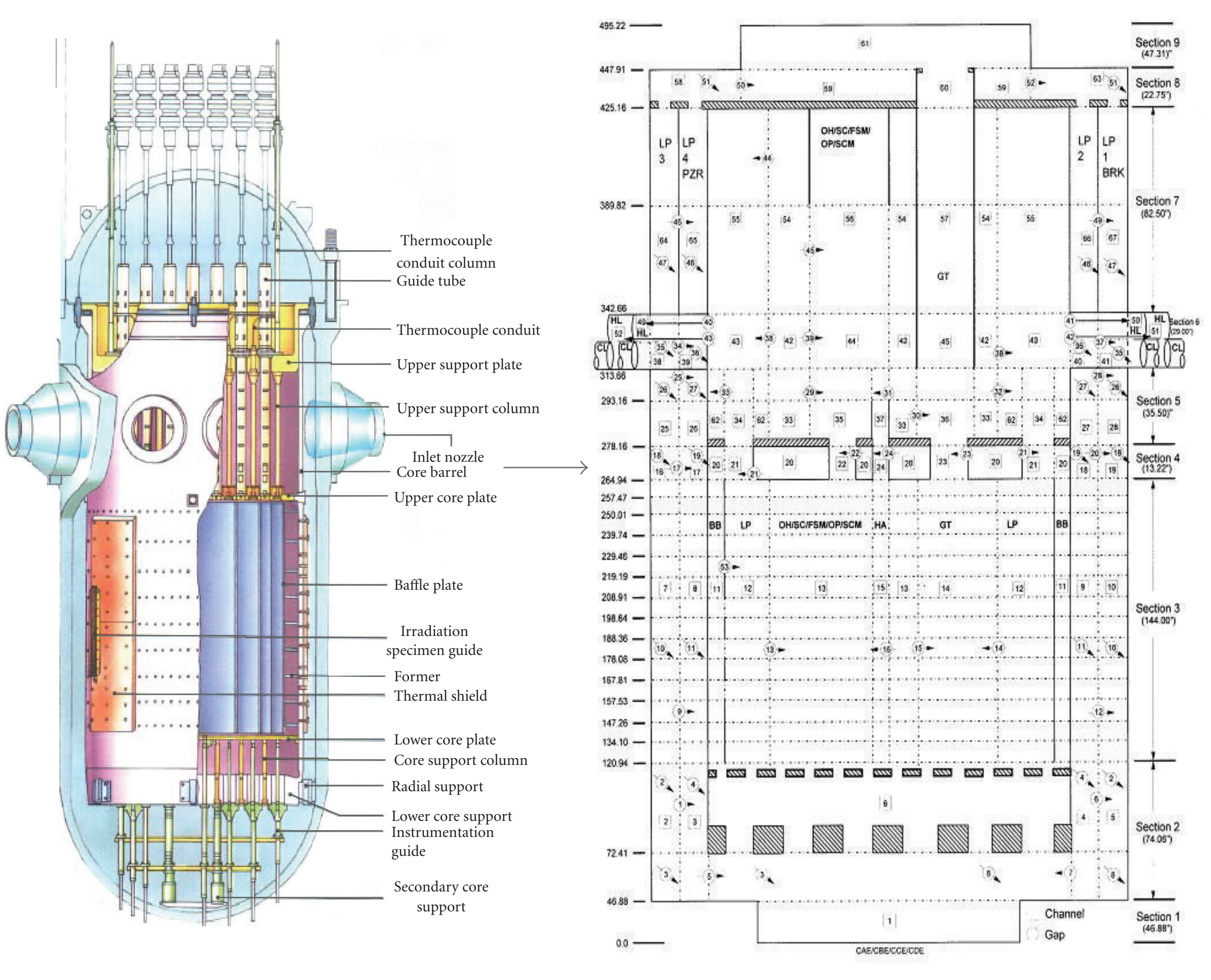

Figure 3: WCOBRA/TRAC typical noding of the reactor vessel.

procedures, consistency with the code assessment conclusion is ensured and "user effects" are virtually eliminated.

4. REVIEW BIASES AND UNCERTAINTIES

The Westinghouse methodology identified more than 30 important uncertainty contributors, as shown in Table 1. The list in Table 1 applies to all the standard Westinghouse 2-, 3- and 4-loop PWR. For the 2-Loop UPI, some additional uncertainty parameters were considered with regard to the upper plenum hydraulics (Takeuchi et al. [5]).

Table 1 is a substantially larger than the list developed in the CSAU demonstration. This fact does not indicate a aw in the CSAU methodology itself, but is indicative of the need to apply the PIRT process thoroughly, and not rely totally on the CSAU demonstration.

Note also that there are many other parameters beyond the list in Table 1 which may affect the results. However these are parameters whose sensitivity to the transient results is expected to be very small or negligible. In those circumstances, it is appropriate to consider those parameters at their nominal (expected or midpoint) value without consideration of uncertainty. Typically, this is a good approximation when the variation in the parameter is tightly controlled, such as pressurizer level, or when the sensitivity to the value of the parameter is known to be negligible, such as a small uncertainty in the vessel and loop dimensions or secondary side liquid mass.

For some other parameters, a conservative value may be used when the parameter varies gradually as a function of operating history, such as steam generator tube plugging, or when the value of the parameter at the time of the accident is indeterminate, such as location of the pressurizer relative to the break. A parameter may also be bounded when the sensitivity of the transient results of variations in the parameter is small, such as moderator temperature coefficient, or when the effort to develop and justify a detailed uncertainty treatment was judged to exceed the benefits of doing so, such as containment pressure response.

The Westinghouse methodology considers the distinction between global and local variables. Each LOCA transient analysis is divided in two parts as follows.

(1) Predict the nominal behavior of fuel rods in the high power fuel assembly, as a result of variations in global variables. Global variables are defined as those variables which affect the overall system thermal-hydraulic transient response. By nominal we mean the predicted fuel behavior when local variables (see below) are at their as-coded or best-estimate value.

(2) For a given reactor, coolant system response, and nominal (see definition above) hot assembly behavior, predict the behavior of the hot rod as a result of variations in local, or hot spot, variables. Local variables affect the hot spot response, but have a negligible effect on the overall system thermal hydraulics, which allows us to consider their impact only at the local level.

Variables 24 to 37, for example, pertain to the second category.

Most of the uncertainties in Table 1 with only few exceptions are explicitly treated and propagated during the uncertainty analysis. The only exception is in the treatment of the ECCS bypass and the entrainment into the steam generators where the mild conservative bias observed during the code assessment against full-scale data is accepted. Another example of accepted conservative bias is the reflood heat transfer coefficient in the core during the initial insurge of water at the end of refill. The heat transfer is limited to a maximum value during reflood due to the lack of data at high reflood rates.

For each contributor in Table 1, the range over which the variable was expected to deviate from the nominal (i.e., as input or as coded value) was quantified using SETs and IETs data or plant operation data. The end result is a probability distribution function for each of the uncertainty parameters. For the plant operating conditions, this quantification was relatively straightforward. For example, the average power in the hot rod is constantly monitored during plant operation. However, uncertainties are introduced by the measurement and the software used in the control room to convert the raw measurement to a linear heat rate. These uncertainties have been thoroughly quantified by Westinghouse in actual reactors.

For thermal-hydraulic models, the analysis was more difficult. Each process has to be described in term of a single modeling variable. For example, the ratio of the measured to WCOBRA/TRAC predicted critical flow rate (CD) is identified as the modeling variable to describe the ability of the code to predict critical flow. The uncertainty probability distribution function of the modeling variable (CD in this case) is determined by generating a scatter plot obtained from the simulation of several critical flow experiments with WCOBRA/TRAC. Then, it had to be demonstrated that the model used to simulate each specific process was sufficiently correct so as not to introduce significant bias or scatter which did not reflect true uncertainty. This was required because the scatter plot used to quantify the uncertainty must not be dominated.

For some parameters. the probability distribution functions were approximated by normal distributions; for other parameters, an "actual" distribution was used. In some cases, a uniform distribution is assumed if the information was insufficient to characterize a more appropriate distribution.

Note also that a detailed compensating error analysis was performed to investigate the interaction of various models and identify situations where an apparently good prediction is due to offsetting mispredictions. The analysis was reviewed by the NRC in order to assess the code's ability to correctly predict all the dominant physical processes during the transient.

5. REVIEW UNCERTAINTY ANALYSIS METHODS: FROM RESPONSE SURFACE TECHNIQUES TO APPLICATION OF NONPARAMETRIC STATISTICS

Element (3) of the CSAU roadmap discusses how uncertainties are combined and propagated throughout the transient. In Element (2), probability distribution functions have been obtained for all uncertainty parameters (about 40 in

Table 1: PWR uncertainty contributors.

	(a) Plant initial uid conditions
1	RCS average uid temperature
2	RCS pressure
3	Accumulator uid temperature
4	Accumulator pressure
5	Accumulator volume
6	Safety injection temperature
7	Accumulator line resistance
	(b) Plant initial core power distribution
8	Core power calorimetric uncertainty
9	Decay heat uncertainties
10	Gamma redistribution
11	Nominal hot assembly peaking factor
12	Nominal hot assembly average relative power
13	Average relative power, lower third of core
14	Average relative power, middle third of core
15	Average relative power, outer edge of core
16	Time in cycle
	(c) Thermal-hydraulic physical models
17	Break type (cold leg split or guillotine)
18	Break area (for split breaks)
19	Critical flow modeling (CD)
20	Broken loop resistance (pumps and other loop resistances)
21	Condensation modeling
22	ECC bypass entrainment and steam binding
23	Effect of nitrogen injection
	(d) Hot rod physical models
24	Local hot spot peaking factor
25	Fuel conductivity
26	Gap heat transfer coefficient
27	Fuel conductivity after burst
28	Fuel density after burst (fuel relocation)
29	Cladding reaction rate
30	Rod internal pressure
31	Burst temperature
32	Burst strain
33	Blowdown heat-up heat transfer coefficient
34	Blowdown cooling heat transfer coefficient
35	Refill heat transfer coefficient
36	Reflood heat transfer coefficient
37	Minimum film boiling temperature

the Westinghouse methodology). The objective of the uncertainty analysis is to quantify the contributions or better the combined effects of all uncertainties to the PCT (or LMO and CWO) from the various sources. The exact solution of the problem would require to examine all the possible interactions among these parameters.

For example, let us assume a simple problem where there are only two parameters X1 and X2. For each of those parameters there are only three discrete value X1(1), X1(2), and so forth. with a probability of occurrence associated to each value, say P11, P12, and so forth. The exact solution to the problem would require to develop an event and outcomes table which include 9 possible events, 9 outcomes (9 PCT values). The resulting PCT distribution is obtained by arranging the 9 PCT values into bins and developing a histogram. The 95th percentile (probability) PCT is obtained counting the number of occurrence in each bin until 95% of all occurrences have been counted.

Clearly the problem is much more complicated than the example. There are about 40 uncertainty parameters and a continuous probability distribution function (PDF) is associated to each parameter. This leads to an infinite number of possibilities and the problem cannot be solved exactly but the solution needs to be approximated to a certain degree. Several approaches have been proposed over the years and the actual implementation of these methods in the industrial application evolved over the last decade. An overview of various methods is provided in the next sections.

5.1. *Response surface method*

A response surface method was suggested by the TGP in an effort to demonstrate that practical methods could be developed within the CSAU framework which would be acceptable by the NRC. Data points are generated by running the code with specific input variables to perform parametric studies on selected uncertainty contributors. Then response surfaces are fit calculation to these data points. The response surfaces are treated as a "surrogate" of the code which reflects the functionality between PCT and the uncertainty attributes. Finally, these response surfaces are used in a Monte Carlo simulation to generate the output distribution (PCT PDF, e.g.).

An advantage of this approach is that the generation of response surfaces requires a well organized matrix of calculations in which single and multiple effects are evaluated. These calculations allow the analyst to understand how each important contributor affects the PCT.

On the other hand, the actual implementation is not as straightforward. The uncertainty contributors have to be grouped together to limit the size of the run matrix which is a strong function of the number of parameters ranged in the uncertainty analysis. At the same time, it is important to ensure that the run matrix or matrices can adequately highlight key interactions.

The first Westinghouse realistic LOCA methodology (Young et al., 1998 [3]) was based on the use of response surfaces. A list of assumptions was made to solve the problem and they are highlighted in the following. *The first main assumption* was to divide the problem into two parts.

(1) Predict the overall reactor response and the *nominal* thermal-hydraulic condition in the high power fuel assembly, as a result of variations in "global" variables. By *nominal* we refer to the predicted fuel behavior when the local variables (24–37 in Table 1) are set at their as coded "best-estimate" values.

(2) For a given reactor response and nominal hot assembly condition, predict the probability distribution of the hot rod behavior as a result of variations in "local" variables.

Step (1) is based on WCOBRA/TRAC simulation while step (2) is based on local evaluation performance with a one-dimensional conduction code called HOTSPOT. For each WCOBRA/TRAC run, the effect of the local uncertainties is collapsed to a probability distribution by performing a large number (1000) of repeated cladding temperature (HOTSPOT) calculations or trials in which the different values of the local variables are randomly sampled from their respective distributions like in a Monte Carlo simulation. The process is depicted in Figure 4.

It is noted that the HOTSPOT probability distribution is a function of the PCT. For example, an uncertainty on the oxidation reaction will have more effect if the clad temperature is high. In other words, the probability distribution is a function of the "global" thermal hydraulic response. The "local" probability distribution is therefore a conditional probability on the "global" outcome probability.

The segregation of some of the variables into the "local" category reduces the problem somewhat, but the runs matrix required to resolve the effect of the remaining global variables would still be too large.

The second main assumption was that the global parameters can be divided in groups. PCT contributions from each group are assumed independent and can be superimposed. The groups identified were as follows.

(1) Initial condition variables (1–7).
(2) Initial core power distribution (8–16).
(3) Physical model and processes parameters (17–23).

The variables are grouped with the justification that some interactions between variables are more important than others. In particular, interactions between variables in different groups (e.g., the fluid average temperature in Group (1) and the nominal hot assembly peaking factor in Group (2) are considered second-order relative to the interaction within group.

Within each group, some of the parameters were then statistically combined into others as "augmentation" factors and some were simply bounded and removed from the uncertainty analysis. The uncertainty of some other parameters were statistically "collapsed." For example, it was shown that the contribution of the initial condition variables could be combined in a single normal distribution (*third main assumption*). At the end of this process, it was shown that the required WCOBRA/TRAC run matrix contains

(1) 10 initial conditions runs;
(2) 15 power distribution runs;
(3) 14 global model runs;
(4) 3-4 split break runs to determine limiting break area;
(5) 8 additional superposition runs.

The last 8 runs were added to correct for the superposition assumption (second assumption). The run matrix of the additional "superposition step" is defined by combining different values of initial conditions, power distributions, and global models. A bias line is determined based on linear regression from the results of the superposition runs. The bias line correlates the PCT_S obtained from the superposition and the PCT_L predicted from the response surfaces and assuming the linear superposition for a given set of parameters. The end results are an additional PCT penalty which is intended to bound the effect of the nonlinear behavior.

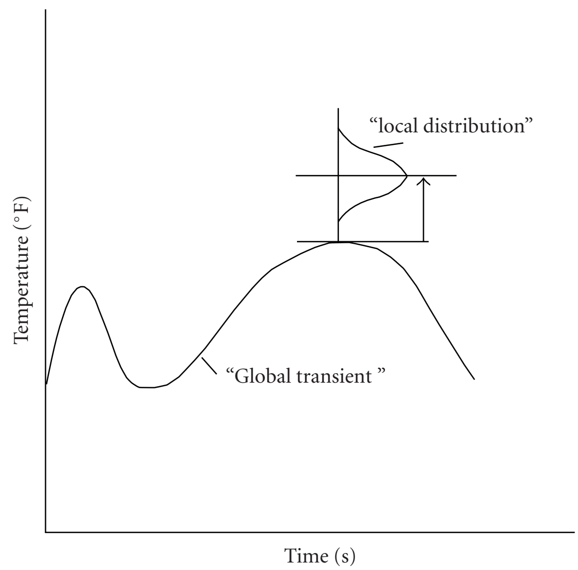

Figure 4: Relationship of local hot spot distribution to global prediction.

With all these assumptions, the problem is reduced to a manageable size with a run matrix of the order of 50 WCOBRA/TRAC simulations.

A criticism on the use of response surfaces is that polynomials could not pick up discontinuities in results or properly identify cliff effects or bifurcations in the results. On the other hand, experience confirms that, at least for large break LOCA, the output is well behaved over a wide range of input values and the response surface seems ideally suited for capturing the local maxima which can occur over the range of variation.

5.2. Direct Monte Carlo method

The problem could be solved (approximated) with a direct Monte Carlo method. The implementation of the method is straightforward and it simply requires to sample the input distributions n times, then use the computer code *directly* to generate n outputs which are used to estimate the actual distribution. The issue is to define how many runs are required to accurately define the distribution of the outcome (PCT, e.g.).

Several years ago, this approach was considered totally impractical due to the number of calculations involved (in the order of thousands). This may not be true today, however. While there are still several issues to resolve with this approach, particularly the number of calculations required to adequately represent the output distribution and extract knowledge about the importance and effect of each contributor, this approach can be considered today.

5.3. Data-based code uncertainty method

A third approach for estimating uncertainty in the PCT prediction due to uncertainties in the thermal-hydraulic models is to compare the computer code to many tests which simulate conditions in a PWR, which result in a measured PCT. Note that this step was also taken in the CSAU demonstration, but the results were not used directly.

The code bias and uncertainty are then determined directly from a PCT scatter plot. The advantage of this approach is that it effectively encompasses all potential contributors to uncertainty. The disadvantages are that the individual contributors cannot be separated, and the propagation of the dominant contributors at full scale is not adequately represented in the data base (e.g., most tests producing a PCT are single effect tests which do not combine the effects of blowdown and reflood).

5.4. Nonparametric statistics method

These methods derive from direct Monte Carlo methods. However, instead of attempting to obtain information with regard to underneath probability distribution function (PDF) of the measure (say PCT), the PDF is ignored (distribution-free) and nonparametric statistics is used to determine a bounding value of the population with a given confidence level.

These alternative methods have been proposed in recent years and started to be applied in realistic calculations of LOCA/ECCS analysis (Wickett Eds et al. [34]). Although, there are some conceptual similarities, most of these methods, started to be employed in Europe in the late 90s (Glaeser et al. [31–33]).

More recently in the US, both AREVA-NP (in 2003) (Martin and O'Dell [35]) and Westinghouse (in 2004) (Nissley et al. [12]) have developed NRC-licensed best-estimate LOCA evaluation models based on the use of these methods. Other applications in the industry are the extended statistical method (ESM) by AREVA/EDF in France (Sauvage and Keldenich [36]) and the GE application to non-LOCA events (Bolger et al. [37]). While all of these implementations utilized essentially the same technique to combine the uncertainties, there are fundamental differences with regard to the interpretation of how these calculation results are used to satisfy the regulatory acceptance criteria.

The nonparametric statistical sampling technique is sometimes referred to as "distribution-free." It is possible to determine the tolerance limits from unknown distributions by randomly sampling the character in question. The consideration of nonparametric tolerance limits was originally presented by Wilks [38]. Wilks study showed that the proportion of the population between two order statistics from a random sample is independent of the population sampled, it is only a function of the particular order statistics chosen. Using the well-known Wilks formula, one can determine the sample size for a desired population proportion at a given tolerance interval. Let us say that we are interested in determining a bounding value of the peak clad temperature (95th percentile ($\gamma = 0.95$)) with 95% confidence level ($\beta = 0.95$). The sample size (i.e., the number of computer runs required) is determined solving the following equation:

$$\beta = (1-\gamma)^N \tag{1}$$

By substituting $\gamma = 0.95$ and $\beta = 0.95$, the number of computer runs, N is found to be 59. In this technique, all the uncertainty parameters are sampled simultaneously in each run similarly to the direct Monte Carlo method discussed in Section 5.2. The method is essentially a crude Monte Carlo simulation used with the minimum trial number to stabilize the "estimator."

Results are then ranked from highest PCT to lowest, rank 1 provides a bounding estimate of the 95th percentile PCT with 95% confidence level.

Beside the PCT, the 10 CFR 50.46 acceptance criteria to be satisfied include also the estimated local maximum clad oxidation (LMO), which needs to be less than 17%, and the estimated value of core wide oxidation (CWO), which needs to be less than 1%.

A rigorous interpretation of the regulations would require the formulation of a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of the three attributes contained in the 10 CFR 50.46 (PCT, LMO, and CWO). The singular statement of uncertainty chosen in this case would be based on a 95% tolerance interval with a 95% confidence level for each of the 10 CFR 50.46 criteria, that is, PCT, LMO, and CWO.

According to Guba et al. [39], this required the extension of the sample size beyond the 59 runs which are only sufficient if one outcome is measured from the sample. A more general theory, which applies to the case where more than one outcome is considered from the sample, is discussed in Guba 2003 paper which provides a more general formula applicable to one-sided populations with multiple outcomes ($P > 1$). The number of runs can be found solving the following equation for N:

$$\beta = \sum_{j=0}^{N-p} \frac{N!}{(N-j)!j!}\gamma^j (1-\gamma)^{N-j}. \tag{2}$$

By substituting $\gamma = 0.95$ and $\beta = 0.95$, and $p = 3$, the number of computer runs, N is found equal to 124. This method was recently implemented in the Westinghouse realistic large break LOCA evaluation model, also referred to as "Automated Statistical TReatment of Uncertainty Method" (ASTRUM) (Nissley et al. [12]). The ASTRUM evaluation model and its approach to the treatment of uncertainties were approved by the US NRC in November 2004.

The implementation or interpretation of order statistics in safety analysis is not fully consistent within the industry. This has led to an extensive public debate among regulators and researchers which can be found in the open literature (Makai and Pál [40], Wallis et al. [41–43], Orechwa [44, 45] and Nutt and Wallis [46]). The focus of this debate has been mostly on the minimum number of runs (sample size) required to satisfy the LOCA licensing criteria (10 CFR 50.46).

Westinghouse strategy was to take the most generic and robust approach to the issue and minimize licensing risks to its customers. Westinghouse position is that there are three criteria that need to be satisfied simultaneously with a singular statistical statement in the form of 95/95. Further, no assumption is made with regard to degree of correlation between the three parameters (PCT, LMO, and CWO) which are measured against the criteria. Based on these assumptions, the sample size is obtained from the Guba and Makai equations (2) and results in 124 calculations.

The maximum values for PCT, LMO, and CWO are extracted from the sample and used as bounding estimators of the 95th percentile for all three quantities with 95% confidence level. The correct interpretation of the results thus obtained is as follows: there is at least a 95% confidence that the limiting PCT, LMO, and CWO from the sample exceed the "true" 95th percentile).

In general, this approach has been considered (overly) conservative, and various authors have suggested that a reduced number of runs would be sufficient compared to what is considered in the Westinghouse methodology. For instance, another approach assumes that while nothing is known relative to the output variable PDF, a strong correlation may exist between the output variables. For example, typically the local maximum oxidation is a strong function of the PCT. However, this approach may require that such a correlation being demonstrated and quantified for the specific analysis.

Both methods are considered acceptable, and each presents advantages and disadvantages. Westinghouse feels that the use of the most generic and robust approach simplifies the licensing and approval process, without requiring plant specific verifications relative to the degree of correlation between the variables or the dominant nature of one of the three criteria. Additionally oxidation is a function of clad temperature and associated time history, not merely of peak cladding temperature. Westinghouse analysis has shown that while a high degree of correlation between PCT and LMO exists, this is plant specific and a generic statement of perfect correlation can not be supported.

An alternative approach was outlined in recent papers from Wallis [43]. Wallis concluded that no matter what, there is only one "output" of interest from a safety analysis, and that is whether the regulatory criteria that apply to the specific transient under consideration are verified. Considering the application to a LOCA analysis, the question that Wallis therefore wants to address is "how many computer code runs are necessary to guarantee, at a 95% confidence, that there is a 95% probability that a LOCA will result in a PCT < 2200F, an LMO < 17%, and CWO < 1%?" The Wallis answer is that if 59 runs are performed, all resulting in an acceptable result (i.e., PCT < 2200F, an LMO < 17%, and CWO < 1%), then a positive answer to the above question can be provided.

The Wallis approach combines PCT, LMO, and CWO into a "single output." The criteria evaluation process is absorbed into the "black box" and simply gives a binary output if succeeded or failed to pass the requirement (compliance with the ECCS design acceptance criteria). Wallis answers a simple "logical" question as depicted in Figure 5 which was extracted directly from his paper (Wallis [43]).

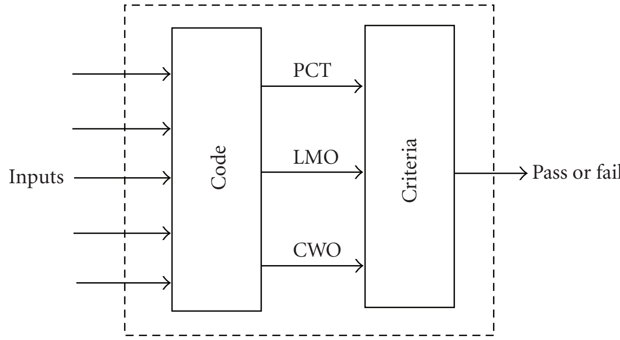

Figure 5: Wallis Formulation.

Wallis's answer is correct in the context of the question as posed, but has in our opinion some limitation in the real application of the method to nuclear safety. In particular, the sample size (number of runs) derived following Wallis's formulation is not sufficient to make a singular statement (at 95/95 level) on the margin that is actually available in the plant design for each of the three criteria. In fact, while there is a 95% confidence that the 95th PCT, LMO, and CWO would be lower than the regulatory limits, the analyst cannot make an estimation (at a 95% confidence level) on how much margin is actually available with respect to the three criteria considered, without decreasing the confidence level or recurring to argumentations based on the correlation between oxidation and PCT.

The quantification and tracking of the margin is most often requested by both the plant operator and the regulator, and the Westinghouse approach (ASTRUM) was sensitive to this issue. More specifically, tracking of PCT margin is a regulatory requirement of 10 CFR 50.46 and cannot be well supported without a quantification of the margin available from the analysis of record.

Further insights on the robustness of the statistical method employed in the current Westinghouse realistic large break LOCA methodology (ASTRUM) are provided in a previous paper (Frepoli and Oriani [13]).

As far as the actual implementation is concerned, ASTRUM evaluation model was grandfathered to the original methodology which was approved in 1996. The extension mainly focused on replacing the method which is used to combine the uncertainties, from the response surface technique to a direct Monte Carlo sampling method. The code (WCOBRA/TRAC) was essen tially unchanged and more the uncertainty parameters were retained with the original probability distribution functions.

One main advantage of ASTRUM is that the number of runs (sample size) is fixed (124 runs) and it is independent on the number of uncertainty attributes considered in the sampling process. As a result, few additional uncertainty parameters were directly sampled instead of choosing the bounding approach considered in the 1996 version of the methodology. To mention some of these new parameters sampled in the procedure: (1) time in cycle on which the postulated LOCA event is predicted to occur; (2) break type (a double ended guillotine or a split); (3) break size for a split break.

The distinction between local and global variables developed in the 1996 version of the methodology was retained in ASTRUM for convenience. However in ASTRUM only a single HOTSPOT calculation is executed downstream a WCOBRA/TRAC, instead of 1000 HOTSPOT runs as in the 1996 methodology. The HOTSPOT calculation is now a single calculation where the local uncertainties coefficients are set at their biased values as selected by random sampling from their respective distributions. This procedure is required to be consistent with the Monte Carlo approach, where a random single-value uncertainty parameter is randomly sampled from the respective distributions for each simulation, which is composed by a WCOBRA/TRAC and a HOTSPOT calculation. There is no need to obtain the "local distribution" depicted in Figure 4 in this case, but simply a random local case within that local distribution, one for each WCOBRA/TRAC run.

6. SAMPLE ANALYSIS RESULTS

Since its original approval in 1996, Westinghouse best-estimate large break LOCA methodology has been applied to perform safety analysis for several PWRs both in the USA and outside. Currently in the US, 24 plants are licensed or analyzed with Westinghouse 1996 and 1999 (upper plenum injection) methodologies and more than 10 plants have been analyzed with the most recent ASTRUM evaluation model which was approved by the NRC in late 2004.

A best-estimate LOCA safety analysis is an engineering project which encompasses several activities. A flow chart is illustrated in Figure 6. Data is collected and compiled in an input model which describes the plant. ASTRUM represents the central phase where uncertainties are combined as discussed in Section 5.4.

Sample ASTRUM analysis results are presented for a typical Westinghouse 3-loop PWR. Other results can be found in the literature (Frepoli [7, 9–11]).

Results for a typical 3-Loop PWR are shown in Figure 7. Figure 7 is a scatter plot which shows the effect of the effective break area on the final PCT. The effective break area is defined by multiplying the discharge coefficient (CD) with the sample value of the break area (FA), normalized to the cold leg cross sectional area. Note that the break area is ranged only for the split breaks (SPLIT), whereas CD is ranged for both split and double-ended-guillotine-cold-leg (DEGCL) breaks. This creates a region in the FAxCD space where both types of break can be found.

Figure 7 shows that the limiting PCT case is a double-ended-guillotine-cold-leg break transient with a near nominal discharge coefficient CD. It is noted that the limiting case with respect to local maximum oxidation (LMO) has rank 2 in terms of PCT and is SPLIT case with a lower effective break area. The LMO case can be easily spotted in the scattered plot of Figure 7, since the PCT is relatively higher than other cases with similar value of effective break area.

Table 2: Sample results of various BELOCA analyses. (Comparison between the 1996 EM and 2004 ASTRUM EM)

Representative plant analysis	1996 EM CQD	2004 EM ASTRUM
2-loop with UPI	PCT = 2087°F LMO < 17% CWO < 1%	PCT = 1870°F LMO = 3.4% CWO $\ll$ 0.3% (18% Power Uprate)
3-loop	PCT = 2050°F LMO = 12% CWO = 0.8%	PCT = 1836°F LMO = 2.9% CWO = 0.03%
4-loop	PCT = 2125°F LMO = 13% CWO = 0.9%	PCT = 1967°F LMO = 2.4% CWO $\ll$ 0.4%

Figure 8 shows the degree of correlation between the local maximum oxidation and PCT for the various runs. While the correlation degree is high, the figure shows that the maximum LMO case does not necessarily coincide with the maximum PCT case.

Figure 9 shows the clad temperature for the ranked top 10 PCT cases. The limiting PCT case and LMO cases are shown in red and green, respectively. It is noted that LMO case is reached in transient which was affected by delay quench. Although the peak clad temperature is lower than the limiting PCT case, more oxidation was occurred in the second case as high temperature were sustained for a longer period of time. The limiting case in term of core-wide oxidation (CWO) had rank 12 in terms of PCT.

Since the limiting PCT, LMO, and CWO values from the run matrix (124 cases) were below the 10 CFR 50.46 limits, a statement can be made were 95th percentile PCT, LMO, and CWO populations are bounded by the limiting values with a 95% confidence level.

Other sample results obtained with both the 1996 methodology (response surfaces) and 2004 ASTRUM (nonparametric) are shown in Table 2. Note that for similar plant ASTRUM provided at least 150 F in additional PCT margin and significant more margins in term of oxidation.

7. SUMMARY AND CONCLUSIONS

Since the 1988 amendment of the 10 CFR 50.46 rule which allowed the use of realistic physical models to analyze loss-of-coolant accidents (LOCA), Westinghouse has been continuously developing and applying its realistic LOCA methodology for the purpose of safety analysis in nuclear power plants. The first version methodology was approved by the NRC in 1996 after an extensive review by the NRC and ACRS.

An overview of the methodology, starting from the thermal-hydraulic code WCOBRA/TRAC and the development of its biases and uncertainties was provided. The paper illustrated that a key step in any best-estimate or realistic analysis is the process selected to combine those uncertainties. Nonparametric order statistics is now the chosen technique to address the issue across the industry. However, the implementation or interpretation of statistics in safety analysis is still not fully consistent within the industry, in

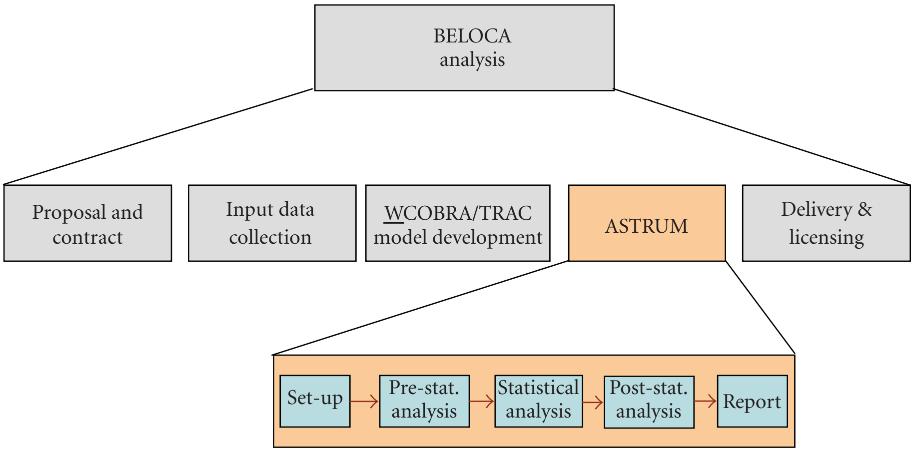

Figure 6: Flow chart of a typical BELOCA analysis.

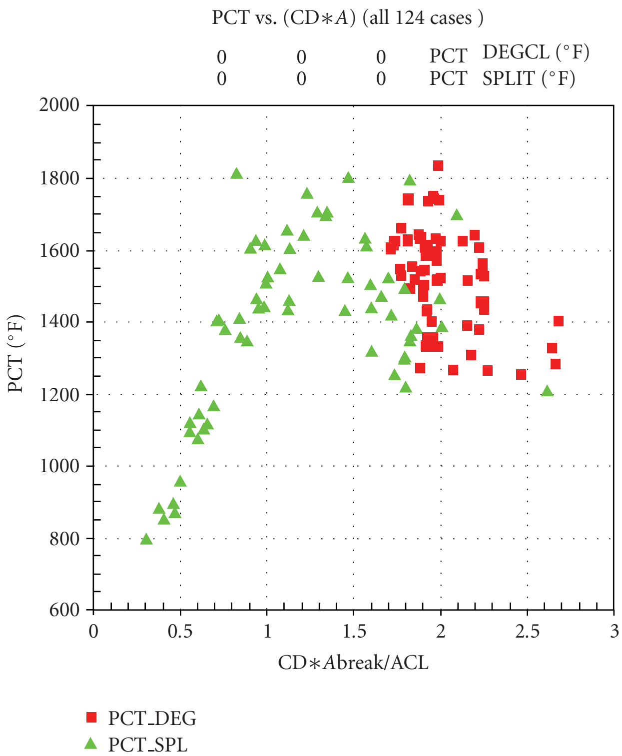

Figure 7: Peak clad temperature (PCT) from the ASTRUM 124 run set.

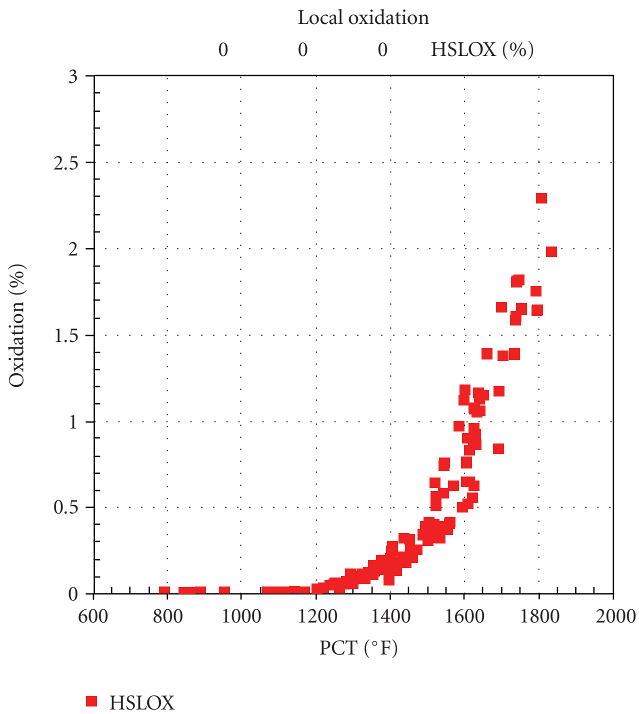

Figure 8: Oxidation and PCT from the ASTRUM 124 run set.

particular with regard to how the analysis satisfies the acceptance criteria set by the regulatory body (i.e., 10 CFR 50.46).

The Westinghouse NRC-approved method (ASTRUM) chooses to follow a rigorous implementation of the order statistics theory, which leads to the execution of 124 simulations within a large break LOCA analysis. This is a fundamentally sound approach which guarantees that a bounding value (at 95% probability) of the 95th percentile for each of the three 10 CFR 50.46 ECCS design acceptance criteria (PCT, LMO and CWO) is obtained. A 95/95 statistical statement on three main ECCS design criteria (10 CFR 50.46) is acceptable by the NRC.

In general, the successful approval of the methodology and several applications to the safety analysis of operating plants that followed is an evidence that the CSAU is indeed a workable roadmap for the development and implementation of realistic methods for safety analysis within the boundaries of the current regulatory environment. Some criticism is still present in the scientific community with regard to CSAU-based methodologies. In particular, concerns with regard to

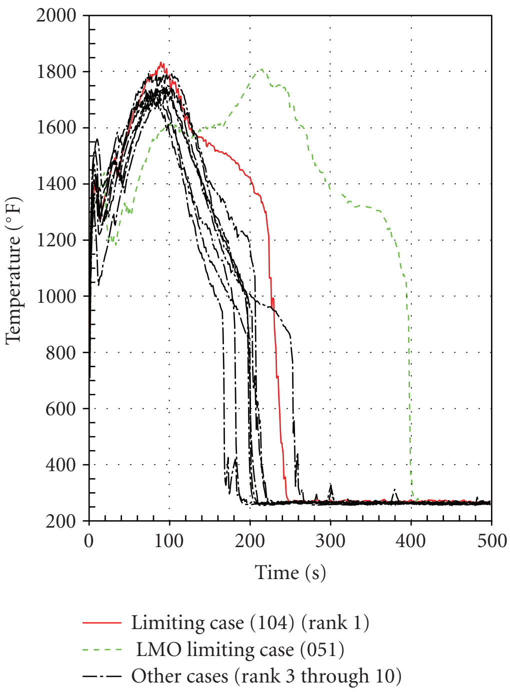

FIGURE 9: Clad temperature traces at the PCT elevation for the top 10 ranked PCT cases in the ASTRUM 124 run set.

the degree of "engineering judgment" within the process are expressed. However a realistic methodology really represents the current state-of-knowledge and the CSAU; and PIRT is a systematic process that allows setting priorities and focus on the most important areas for the purpose of safety analyses. Layers of realistic conservatisism are often added to increase the robustness of the method. Review the methodology by the regulatory bodies, look at the evaluation model in its entirety, and extend well beyond the boundary of what is predicated by the PIRT process. As more information becomes available, information can be used to refine the models. Further improvements typically result in "uncovering" hidden safety margin which may be utilized to improve plant operation performances and economics. Such process prevents technology from being "frozen" in a highly regulated environment and it is in line with risk-informed regulation and defense-in-depth philosophy.

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Review Article

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

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Recommended by Cesare Frepoli

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

1. INTRODUCTION

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

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

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

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

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

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

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

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

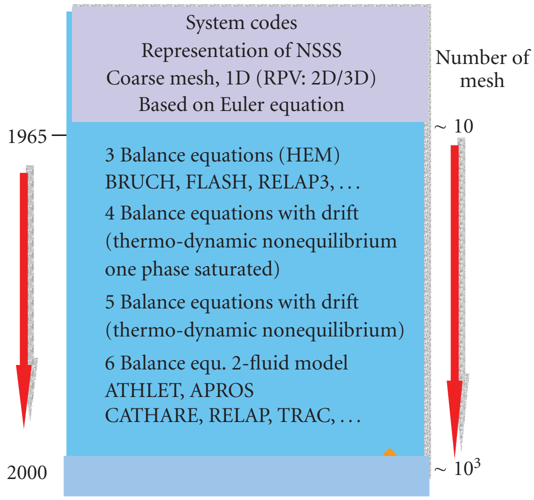

Figure 1: Code development activities in more than three decades.

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

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

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

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

2.1. System nodalization

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

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

2.1.1. Spatial convergence

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

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

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

2.1.2. Mapping of multidimensional effects

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

2.2. Code options: physical model parameters

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

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

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

2.3. Input parameter related to specific system characteristics

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

2.4. Input parameters needed for specific system components

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

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

2.5. Specification of initial and boundary conditions

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

2.6. Specification of state and transport property data

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

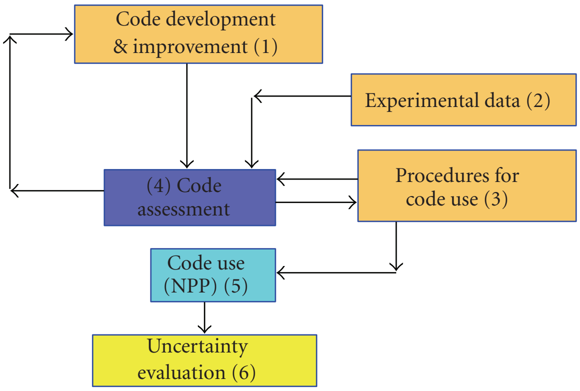

FIGURE 2: A consistent application (development, qualification and application) of a thermalhydraulic system code.

2.7. Selection of parameters determining time step sizes

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

2.8. Code input errors

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

3. QUALIFICATION OF COMPUTATIONAL TOOLS

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

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

3.1. Code qualification

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

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

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

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

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

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

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

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

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

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

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

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

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

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

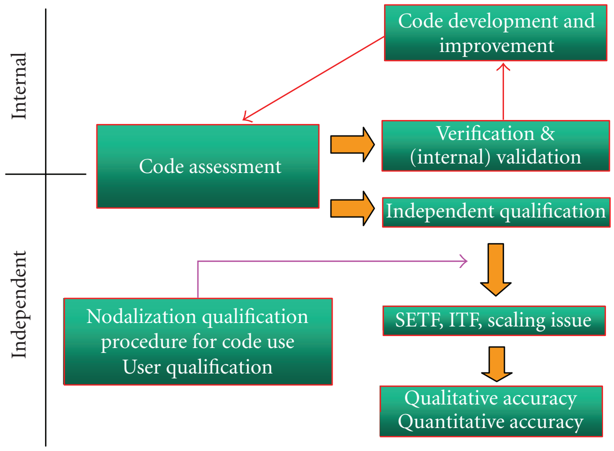

FIGURE 3: Internal and external (independent) code assessment.

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

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

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

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

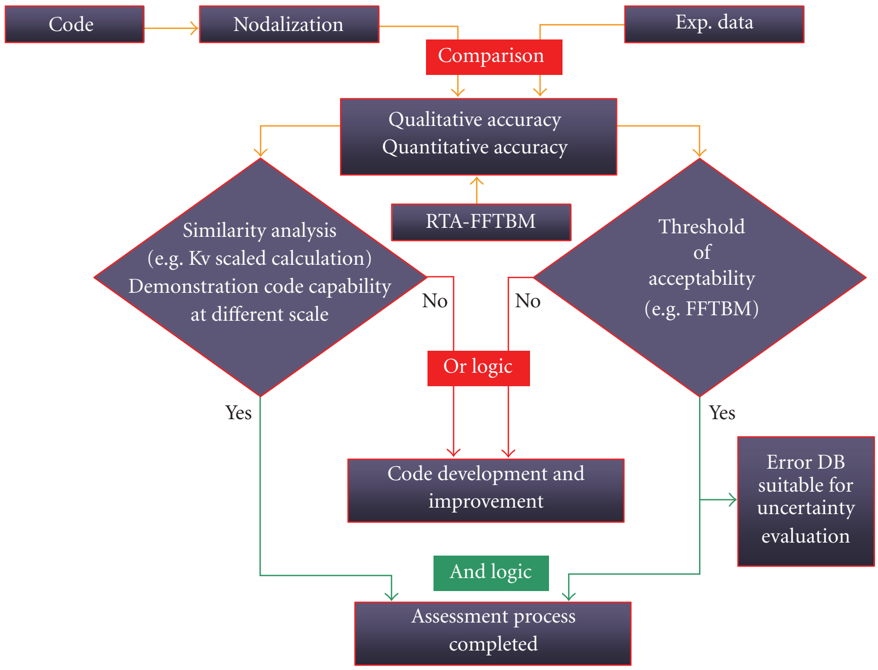

Figure 4: Code independent assessment.

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

Step A

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

Step B

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

Step C

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

Step D

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

Step E

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

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

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

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

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

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

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

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

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

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

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

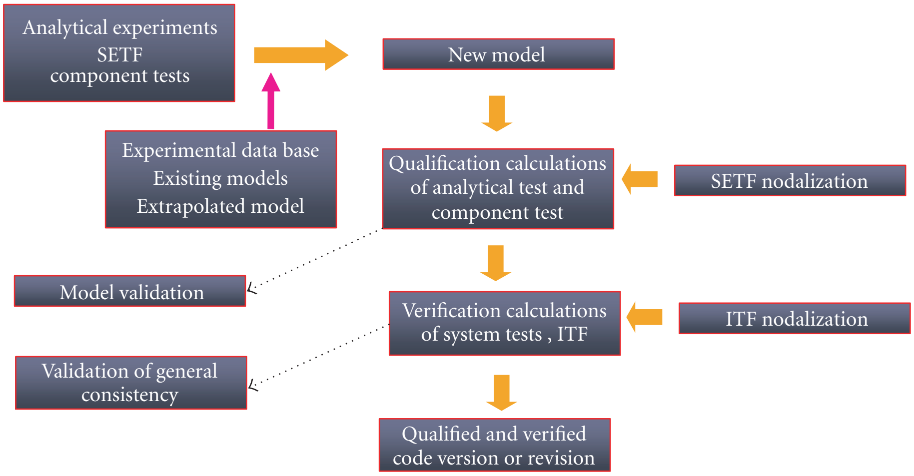

Figure 5: Qualification process of a new revision or a new version of the code.

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

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

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

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

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

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

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

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

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

3.3. Addressing the scaling issue

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

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

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

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

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

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

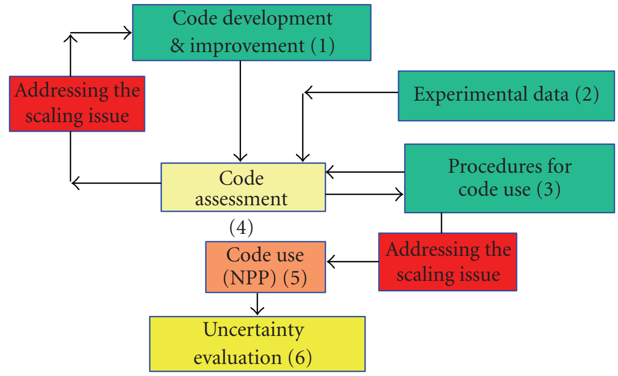

Figure 6: Role of the scaling analysis in the code assessment process.

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

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

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

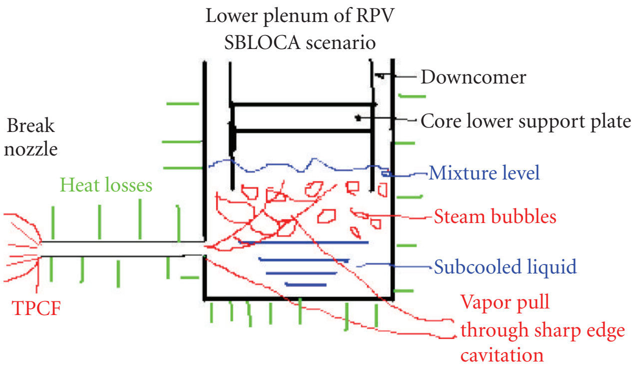

Figure 7: Schematic representation of a two-phase flow condition in a reactor pressure vessel of a facility during an SBLOCA.

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

(a) developing a system code;

(b) qualifying the code against experimental data;

(c) demonstrating that the code-accuracy (i.e., discrepancy between measured and calculated trends) only depends upon boundary initial conditions (BIC) values (within the assigned variation ranges) and is not affected by the scale of concerned ITF;

(d) applying such code to predict the same relevant phenomena that are expected to find in a same experiment (or transient) performed at different scale;

(e) performing NPP Kv-scaled calculation and explaining the discrepancies (if any) between NPP Kv-scaled calculation and measured trends in ITF considering only BIC values and hardware differences (i.e., distortions).

3.4. Nodalization qualification

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

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

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

(1) The data of the reference plant available to the user are typically non exhaustive to reproduce a perfect "schematization" of the reference plant.

(2) From the available data, the user derives an approximated nodalization of the plant reducing the level of detail.

(3) The code capability to reproduce the hardware, the plant systems and the actuation logic of the systems reduce further the level of detail of the nodalization.

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

(1) the code options must be adequate;

(2) the nodalization solutions must be adequate;

(3) some systems components can be tested only during transient conditions (e.g., ECCS that are not involved in the normal operation).

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

Step "a"

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

(1) homogeneous nodalizations;

(2) strict observation of the user guidelines;

(3) standard use of the code options.

Step "b"

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

Step "c"

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

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

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

Step "d"

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

Step "e"

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

Step "f"

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

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

Step "g"

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

Step "h"

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

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

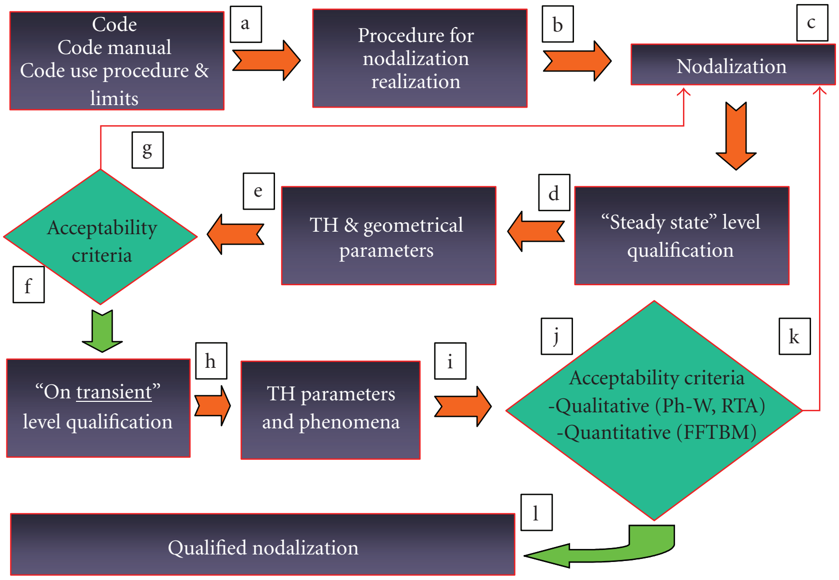

Figure 8: Flow sheet of nodalization qualification procedure.

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

Step "i"

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

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

Step "j"

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

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

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

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

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

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

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

Step "k"

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

Step "l"

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

4. DEVELOPMENT AND USE OF COUPLED COMPUTER CODES

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

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

Table 2

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

5. CONCLUSIONS

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

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

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

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

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

ABBREVIATIONS

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

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

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Research Article

Development Considerations of AREVA NP Inc.'s Realistic LBLOCA Analysis Methodology

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The AREVA NP Inc. realistic large-break loss-of-coolant-accident (LOCA) analysis methodology references the 1988 amended 10 CFR 50.46 allowing best-estimate calculations of emergency core cooling system performance. This methodology conforms to the code scaling, applicability, and uncertainty (CSAU) methodology developed by the Technical Program Group for the United States Nuclear Regulatory Commission in the late 1980s. In addition, several practical considerations were revealed with the move to a production application. This paper describes the methodology development within the CSAU framework and utility objectives, lessons learned, and insight about current LOCA issues.

1. INTRODUCTION

The objective of any methodology for measuring the performance of an emergency core-cooling system (ECCS) during a loss-of-coolant accident (LOCA) is to provide a statement of assurance that the ECCS will preserve fuel integrity. For large-break LOCA analysis, the key measure (among several) is peak cladding temperature (PCT) relative to 2200°F (1200°C). Traditionally, LOCA analyses performed in the U.S. for Nuclear Power Plant design-basis safety analysis were required to comply with the U.S. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix K, a conservative, deterministic approach. Following several research and development advances in two-phase flow and heat transfer phenomena specifically related to the LOCA, regulations were updated in 1988 to allow best-estimate approaches. Several events leading up to the rule change included the close of the 2D/3D program [1] and the development of NUREG-1230, Compendium of ECCS Research [2]. In addition, during the rule-making process, a committee of experts was convened to develop a paradigm for performing best-estimate LOCA evaluations. These experts came from the USNRC, national laboratories, and academia. This Technical Program Group (TPG) produced the code scaling, applicability and uncertainty (CSAU) methodology, which is documented in NUREG-5249 [3]. Today, the CSAU methodology is well known in the LOCA community and many papers have been inspired from both the content and the conclusion developed from that original work. Accompanying NUREG-5249, the USNRC released Regulatory Guide 1.157, best-estimate calculations of emergency core cooling system performance, which provides specific detail describing acceptable best-estimate LOCA methodologies [4].

An AREVA NP predecessor company, Siemens Power Corporation, developed and submitted to the USNRC a best-estimate LBLOCA methodology during the early 1990s; however, the USNRC could not provide resources to support the review for several years. As a consequence, Siemens Power Corporation decided to reinvent this methodology and resubmitted a realistic large-break LOCA (RLBLOCA) methodology in August 2001 [5]. In April 2003, AREVA NP received approval of an S-RELAP5-based realistic large-break LOCA methodology from the USNRC [6].

2. EVOLUTION OF BE METHODS SINCE 1988

The development of this methodology is a product of the lessons learned since the 1988 rule change both internal to AREVA NP and by the thermal-hydraulic community at large. This is despite the fact that in May 1990 in a special issue of "Nuclear Engineering and Design" [7], the editor declared "the closure of the large-break LOCA issue." This bold statement did not go unchallenged. In January 1992, a special issue of "*Nuclear* Engineering and Design" [8] was published providing comment and criticism, in the form of "Letters to the Editor," of the existing technical understanding of LOCA, in general, and the CSAU methodology, specifically. Several areas were identified as being incomplete. These can be generally associated in the following categories [9]:

(i) defining "best-estimate methods;"
(ii) merits of engineering judgment;
(iii) methods for the convolution of uncertainty;
(iv) data to quantify uncertainties.

In order to produce an acceptable, usable methodology, resolution of these and other issues was necessary. "Resolution" is, of course, a negotiated condition involving the methodology developers, an applicant, and the regulatory reviewers. Nonetheless, this paper presents insights from AREVA NP's experience in the process from the 1988 rule change until US-NRC approval in 2003.

2.1. Defining "best-estimate" methods

In the context of thermal-hydraulic safety analysis performed to support nuclear power plant operation, no consensus appears to have been established for defining "best estimate." The difficulty stems from the many types of uncertainty contributing to a plant-scale accident scenario. Sources of uncertainty associated with a large-break loss-of-coolant accident (LOCA) analysis begin with that which can be observed—measurable quantities reflecting the design or condition of a system, structure, or component. In this context, "best-estimate" can be simply characterized as a preferred state for which any perturbation is followed by a return to its preferred or "best-estimate" state for the system, structure, or component.

The original problem tackled by the TPG in NUREG-5249 was for a double-ended large-break LOCA at a Westinghouse 4-loop PWR operating at steady-state full power. Several uncertainties associated with this problem were recognized in that reference including those associated with code models, the impact of test facility scaling, epistemic uncertainty resulting in compensating errors, nodalization, and, to a lesser extent, the user effect. On the surface, the TPG appeared to establish a well-defined description; however, even this description, supported by the discussion on uncertainty presented in NUREG-5249, disguises other uncertainties that are much more difficult to quantify and include into a definition for "best-estimate methods." To identify these additional uncertainty contributors, this application statement can be dissected.

Beginning with "double-ended large-break LOCA," this identifies a scenario with a particular break configuration. This vision of the large-break LOCA problem either ignores the spectrum of breaksizes associated with LOCAs or addresses this uncertainty with conservatism. Incorporating conservatism into the definition of "best-estimate methods" appears to undermine the original move to best-estimate methods. One of the primary criticisms of the Appendix K deterministic approach was that certain so-called "conservative" models could result in nonconservative behavior during a simulation. Best-estimate methods certainly should avoid this situation; however, the question of breaksize is just one element of the broader uncertainty category associated with the nature of the initiating event. The communicative nature of the break (i.e., guillotine or longitudinal split), break orientation (i.e., necking for guillotine break and directional nature of split breaks), break location (i.e., cold or hot leg; pressurizer or other loops, attached pipe), and the assumed single failure (a regulatory requirement) also contribute to the initiating event uncertainty.

The descriptor "Westinghouse 4-loop PWR" identifies a plant design; however, the nature of nuclear power plant development is such that even among Westinghouse 4-loop PWRs there can be significant differences. Component choices, such as reactor coolant pumps, steam generators, core/reactor vessel design (i.e., bypass flows, fuel assembly design, upper head design), and containment response features (i.e., sprays, ice, fan coolers, passive structure surface area), represent elements of the design uncertainty. In addition, operational and maintenance history can impact the performance of "equivalent" systems, structures, and components. As a consequence, there are no "identical" plants.

"Operating at steady-state full power" encompasses all uncertainties associated with plant operating state and event response. In analysis space, these are often initial or boundary conditions. A plant's technical specifications and limiting condition of operation define the operational space enveloping acceptable plant states. Frequently, there is significant latitude for "acceptable" states for system variables, including core axial power and fuel burnup that can have a strong influence on the acceptance criteria metrics. The challenge for a "best-estimate" analysis is to balance the value of defining the likely plant state at the time of an accident with the need to support the plant's operational envelope. Dozens of analysis parameters fall into this category.

In recognizing the complexity of the uncertainty problem associated with LOCA safety analysis, the term "best-estimate" as applied to this problem has evolved into "best-estimate plus uncertainty" (BEPU). The problem has always been the management of uncertainty. At the time the CSAU methodology was being developed, a relatively narrow view of uncertainty was necessary because of limitations in computational ability and limited appreciation of advanced statistical methods. This original CSAU view on uncertainty was criticized as being incomplete with relevant contributors to the LOCA safety analysis problem being treated implicitly and, as a consequence, wrong. As such, the conversation moved from BE to BEPU—with the emphasis on uncertainty management.

2.2. The role of engineering judgment

Engineering judgment has always been a necessary part of any engineering task. Engineers, through the expression of their experience, have often applied engineering judgment to make big engineering challenges workable. Confirmation is, of course, necessary when safety is a concern. In developing the CSAU methodology, the TPG formalized this often unappreciated aspect of engineering. Doing so started a debate as to the extent that engineering judgment should play in the LOCA safety analysis problem.

The manifestation of engineering judgment in the CSAU process is the phenomenological identification and ranking table (PIRT). As the name implies, the PIRT reflects qualitative engineering judgment as to the importance of various phenomena relevant to the problem of interest. The intent of the PIRT is to provide a technical basis during the BEPU methodology development process for the many decisions, including the management of uncertainty, required to complete the task.

Resistance to this formalized use of engineering judgment inspired several criticisms, including the following.

(i) Who is qualified to be a part of a PIRT team?

(ii) How do PIRT teams deal with differences of opinion?

(iii) Should uncertainty with the ranking process be incorporated into the PIRT?

(iv) Even after the PIRT is developed, engineering judgment is required to use the results.

(v) How can the absence of knowledge (i.e., unmodeled parameters) be treated in this context?

Despite the initial criticism, the PIRT exercise has found a degree of acceptance. Its foremost value has been in establishing an understanding of the processes and phenomena of interest among a group of peers. Once consensus is achieved, decisions impacting the solution of the task at hand may begin.

In the original CSAU large-break LOCA sample problem, the TPG, applying a PIRT they developed for this problem, established a precedent that the large-break LOCA problem can be well characterized by explicitly addressing a minimum set of very important processes and phenomena. Beyond that set of large-break LOCA contributors, other phenomenological or process parameters were treated as "nominal." This application of engineering judgment has not found universal acceptance for two reasons: (1) there is a lack of consensus of "important" parameters and (2) it ignores traditional licensing measures defined in plant technical specifications and limiting condition of operation.

To satisfy this criticism, the BEPU approach recognizes the value of "realistic conservatism," that is, the explicit treatment of uncertainty by characterizing the uncertainty parameter such that the key output variables are penalized relative to the acceptance criteria. For parameters with low large-break LOCA importance, this may be a trivial distinction; however, as importance increases, scrutiny over that which is proclaimed conservative also increases. Nonetheless, the acceptance of "realistic conservatism" represents a significant departure from the original concept of BE methods; yet, it is absolutely necessary for the complex LOCA analysis problem where engineering judgment is involved.

2.3. Convolution of uncertainty

A constraint, recognized early by the TPG during the development of the CSAU method, stemmed from the application of statistics to convolve parameter uncertainty of several individual large-break LOCA contributors into a single uncertainty statement for PCT. Specifically, the broader the set of uncertainty contributors considered, the more than number of required LOCA simulations grows exponentially. This is the nature of the response surface methods that the TPG considered state-of-the-art for this application. There is no doubt that this practical constraint influenced their acceptance of the relatively small number of large-break LOCA contributors considered in their uncertainty analysis sample problem. Later, Westinghouse would introduce a clever extension to the response surface approach to expand the number of large-break LOCA contributors that could be considered [10].

When introduced in 1989, a few organizations in the international thermal-hydraulic community—in particular, Germany's GRS—recognized that this obvious limitation could be eliminated by considering nonparametric statistical approaches. This counterpoint was not universally appreciated either because there was a lack of understanding or nonacceptance of nonparametric statistics lack of a definitive uncertainty statement. The uncertainty statement from a nonparametric statistical approach is expressed as an inequality characterized with a confidence level.

Today, nonparametric-ordered statistics (e.g., Wilk's method) have become the method of choice. However, consensus with regard to its implementation within regulatory guidelines is still evolving. Current regulation in the U.S. and other countries recognize a multivariant acceptance criterion for large-break LOCA analysis. As a consequence, a debate over the required number of calculations necessary to provide an acceptable uncertainty statement has resulted in several journal articles on the subject [11–15]. Much of this debate is on the semantics used to present the uncertainty statement. Specifically, should the acceptance criterion be measured individually or is it sufficient to consider the outcome of an analysis as a single statement concerning whether the entire acceptance criterion has been satisfied. AREVA NP's position is with the latter.

2.4. Completeness of the experimental database

Driven by the recognized gap in knowledge of LOCA phenomena apparent in the early 1970s that resulted in the early Appendix K rule making, governments around the world invested heavily in experimental programs to rectify this situation. By the late 1980s, a large body of research on many facets of the large-break LOCA problem was completed. Coupled with the CSAU approach for performing BE analysis, was this body of work sufficient to declare the closure of the large-break LOCA problem? Undermining the closure position was the view that so much of the thermal-hydraulic phenomenological database was populated empirically and, as such, there remains much yet to be characterized.

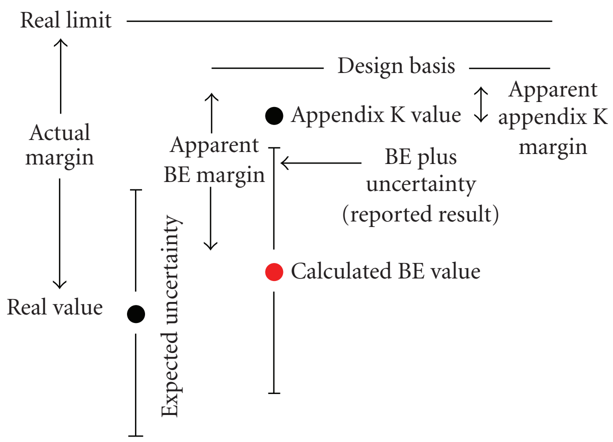

Figure 1: Illustration of plant operating margin.

While statistics appeared to be the answer to the analyst, the experimentalist was saying that in many areas data was insufficient for deriving statistical measures. In addition, the possibility of unknown phenomena or undesirable interplay between competing phenomena made any declaration of closure irresponsible. The TPG's response was simply that a sufficient amount of experimentation focused on both separate and integral effects existed and that uncertainty associated with scale could be determined. In areas this may be large; however, if it turns out that uncertainty is too penalizing, this would be a motivation for new test programs.

2.5. AREVA NP's BEPU paradigm

Constraining factors that can limit a nuclear power plant's efficiency include engineering design limits, equipment operability, and regulatory requirements. The acceptance of BE methods has revealed margin for improving plant operating performance. Figure 1 illustrates this view of the plant operating margin provided by BE methods relative to the traditional Appendix K deterministic methods. Margin is characterized by the separation between the design or the licensing limit and the nominal operating point. With regard to regulatory limits, this is measured by recognized metrics relative to the regulatory acceptance criteria, for example, PCT $<$ 2200°F.

Deterministic methods providea single "analysis of record" that quantifies the acceptance criteria metrics (PCT, total oxidation, and local hydrogen generation).Over the operating history of current generation nuclear power plants, utilities have nearly exhausted the availability of margin provided by this original method and, as a result, the apparent margin is small.

In contrast, BE methods strive to identify the acceptance criteria metrics associated with the real state of the plant. Practical limitations associated with the state of knowledge required to perform analyses force analysts to apply conservatisms that make the calculated BE value bounding of the real state. In addition, the real margin is never realized because the design basis limits reserve margin to cover uncertainties associated with the actual limits.

For the purpose of reporting plant operating performance margin relative to licensing limits, the goal is not to define this margin relative to the actual state; rather, it is to convolve all key phenomenological and process uncertainties to identify the calculated BEPU value—a conservative estimate of margin incorporating realistic models of the physical processes and associated phenomena.

In preparing the AREVA NP large-break LOCA methodology, the challenge of addressing the expectations of Regulatory Guide 1.157 and the CSAU process—balanced with the known criticisms of the CSAU process—moved the AREVA NP methodology development team towards nonparametric statistical methods and the "realistic conservatism" concept of uncertainty management. By taking this step, the focus of the methodology moves towards the resolution of individual uncertainty contributors.

The main advantage of nonparametric statistical methods is that the number of treatable uncertainty contributors is independent of the number of plant calculations. This characteristic provides flexibility during the development process to explicitly address as many or as few analysis contributors as necessary to resolve the outcome of the PIRT. As this is a product of engineering judgment, the uncertainty associated with this exercise can be reduced by explicitly addressing additional analysis contributors. In addition, this methodology characteristic provides the opportunity to incorporate customer requests for the explicit treatment of plant process uncertainty.

For the remainder of this paper, a description is provided of how AREVA NP's RLBLOCA methodology conforms to the basic principles of the CSAU methodology while incorporating realistic conservatisms and nonparametric statistics.

3. RECONCILING AREVA NP's RLBLOCA METHODOLOGY WITH CSAU

The development of AREVA NP's RLBLOCA methodology was primarily an exercise in complying with the main themes of the CSAU methodology. AREVA NP's interpretation of the CSAU approach is that it represents a framework for deriving a quantifiable degree of assurance from a best-estimate analysis tool. This framework, graphically presented in Figure 2, consists of three elements and 14 steps that build on a qualitative understanding of (in this case) the large-break LOCA problem to define the necessary tasks to derive a quantitative solution. Highlighted components in Figure 2 represent steps that overlap with deterministic Appendix K methodologies. The CSAU framework outlines a procedure that leads from the identification and characterization of the dominant phenomena influencing the key acceptance parameter, PCT, to quantify a best-estimate of the consequences of a LBLOCA and its associated uncertainty. As with Appendix-K-derived methodologies, the final result is a calculation that provides a PCT to be measured against the 10 CFR 50.46 acceptance criteria and a statement of total uncertainty associated with that result.

Table 1: AREVA NP's choices for CSAU steps 1, 2, 4 and 5.

PIRT step	AREVA methodology
Specify scenario	Large-break LOCA
Selection NPP	Westinghouse and CE PWRs with cold leg SI
Select frozen code	RODEX3A (fuel performance) S-RELAP5 (RCS and Containment Thermal-Hydraulics)
Provide documentation	[5, 18–24]

3.1. Requirements and code capabilities

The first CSAU element sets a foundation of understanding to guide methodology development. Its emphasis is on defining the problem and capturing a knowledge base that will be used to provide the fundamental technical basis for decisions downstream in the methodology development process. Steps 1, 2, 4, and 5 shown in Figure 2 identify the problem through specification of the event scenario, plant type, computer code and version, and computer code documentation, respectively. Historically, this information represented all that would normally be required for evaluation methodologies (EM) based on 10 CFR 50 Appendix K. Table 1 summarizes the AREVA choices. Of particular note is the primary analysis tool S-RELAP5. S-RELAP5 is a modified version of RELAP5/MOD2 [16] with several updates including:

(i) multidimensional modeling capability (two-dimensional hydrodynamics);
(ii) energy equations modified to better conserve transported energy;
(iii) incorporation of a derivative of the CONTEMPT [17] containment analysis code;
(iv) iterative evaluation for choked junctions;
(v) bankoff CCFL model;
(vi) modeling of noncondensable gases (e.g., nitrogen discharge form accumulators);
(vii) revised two-phase pump degradation based on EPRI data;
(viii) improvements to interphase friction and mass transfer models;
(ix) Sleicher-Rouse used for single-phase vapor heat transfer.

Step 3, identify and rank phenomena, marks a significant departure from traditional evaluation methodology approaches by formulizing engineering judgment to aid both methodology development and regulatory review. This is particularly important given the substantial effort required to develop a CSAU-based methodology. Step 3 acknowledges that plant behavior is not equally influenced by all processes and phenomena that occur during a transient. This provides the basis to reduce the analysis effort to a manageable set of phenomena ranked with respect to their influence or importance on the primary safety criteria (i.e., PCT).

Table 2: Key LBLOCA phenomena identified by AREVA.

PIRT parameters
Heat transfer
Void distribution
Axial power distribution
Entrainment
Spacer effects
Break flow
Cold leg condensation
Interfacial heat transfer
Upper tie plate CCFL
Core multidimensional flow
ECCS bypass
Steam binding
Accumulator nitrogen discharge

The ranking process employed for the AREVA NP RLBLOCA methodology was accomplished primarily through structured discussions among AREVA NP engineers and recognized nuclear safety and thermal-hydraulics experts from industry and academia. The experts assembled for this task had extensive experience in both the experimental and computational areas of nuclear thermal-hydraulics. The PIRT team started with the original LBLOCA PIRT presented by the TPG [4]. This initial PIRT was reviewed by the three external experts, who offered recommendations for the addition or deletion of phenomena from the PIRT and revisions to the ranking of the phenomena based on the evolution of LBLOCA understanding since the publication of the CSAU methodology and lessons learned from early applications of BE methods. Following this review, a peer review was held with the three experts and four additional AREVA NP personnel to derive a final PIRT that incorporated the input from all seven participants. This final PIRT also merited from approximately 300 code sensitivity studies that served as a validation of the engineering judgment statements. The outcome of these meetings was an AREVA NP-proprietary phenomena identification and ranking table (PIRT) for large-break LOCAs that has many similarities with the original TPG large-break LOCA PIRT [4]. AREVA NP identified the PIRT parameters shown in Table 2 as dominant in a large-break LOCA and must be explicitly addressed in a CSAU-based methodology. Following PIRT development nearly 100 unique sensitivity studies were performed to assess consistency between the PIRT and S-RELAP5 large-break LOCA model response. The outcome of those studies served to motivate further code model upgrades and validate PIRT selections.

CSAU, Step 6, serves to establish a computer code's applicability to the analysis problem. This is done by defining a cross reference of phenomena and plant components to the computer code's models and correlations and nodalization capability. With regard to the dominant PIRT parameters, code applicability also must be supported by the documentation provided in Step 5.

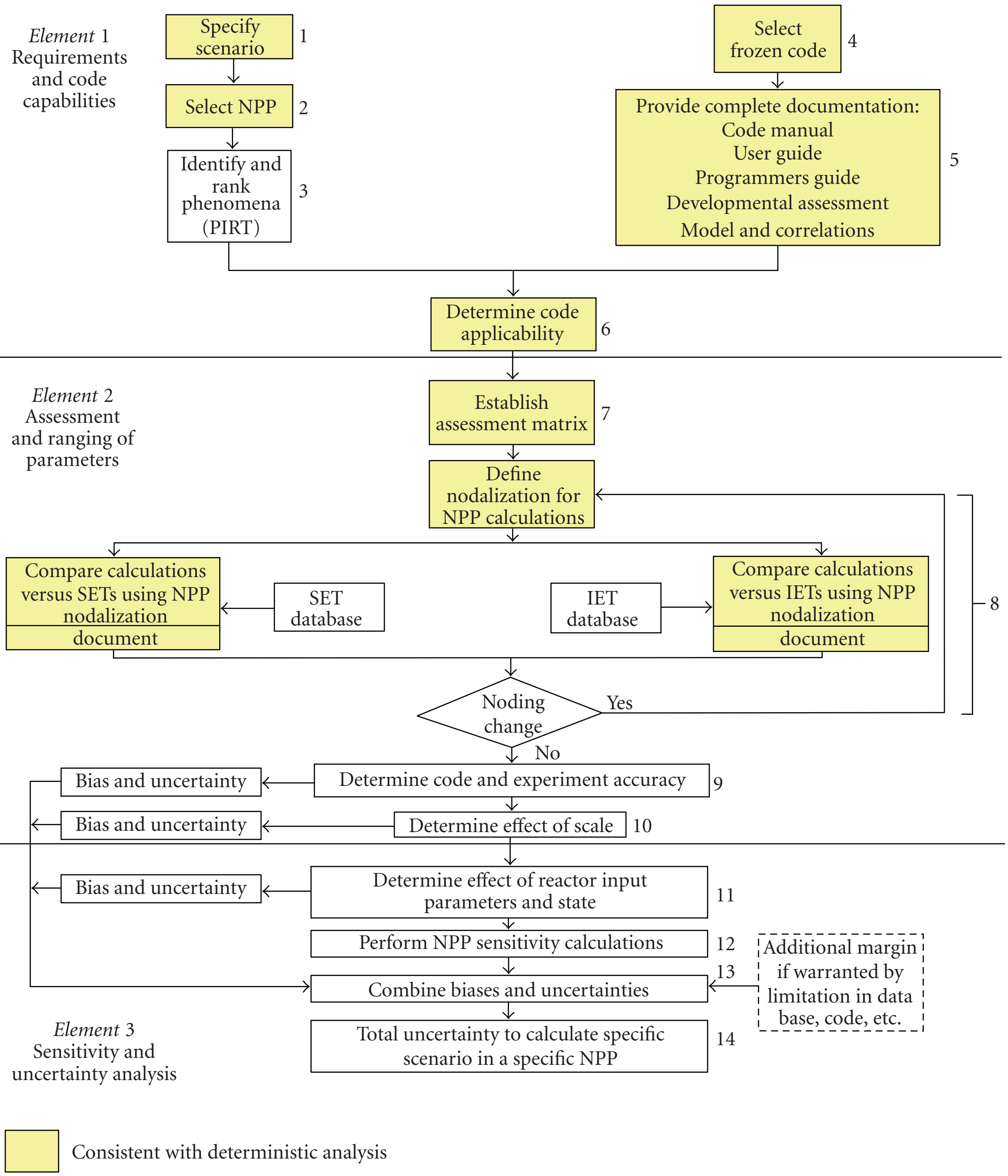

Figure 2: The CSAU methodology framework.

3.2. *Assessment and ranging of parameters*

The second CSAU element establishes the methodology's pedigree to perform a best-estimate analysis. This is done by code-to-data comparisons, sensitivity studies, and uncertainty analysis. It builds from Element 1 that defines a framework for the performance of sensitivity studies and identification of experimental test programs by relevance to the dominant large-break LOCA phenomena. Step 7 defines the code's assessment matrix. Thermal-hydraulic computer codes like S-RELAP5 include a large number of closure-relationships to address the broad spectrum of possible thermal-hydraulic phenomenological processes. For this reason, it is neither practical nor necessary to assess every code model and correlation to support the subset of important phenomena anticipated during a LBLOCA. The PIRT and the subsequent sensitivity studies were used by AREVA NP to identify the most useful experimental programs for code assessment from the rather extensive knowledge base of experiments supporting PWR LOCA phenomena. Proprietary restrictions reduce this set considerably; however, sufficient data remains in the public domain to support qualification of a best-estimate LOCA code for PWR applications. The AREVA NP RLBLOCA assessment matrix is characterized in Table 3, which identifies the test program, the number of specific tests applied to the AREVA NP RLBLOCA assessment matrix, and the primary phenomenon of interest. The particular tests were selected to address the following:

(i) important LOCA phenomena defined in the PIRT;
(ii) nodalization validation (defined in CSAU Step 8);
(iii) code/model scaling (defined in CSAU Step 10);
(iv) verification of no important compensating effects;
(v) establishing a broad range of applicability.

The CSAU methodology acknowledges that system nodalization is similar to any code model or correlation in that code results are sensitive to model permutations. This is addressed in Step 8, nuclear power plant nodalization definition. System nodalization presents an inherent code uncertainty. Unlike code models and correlations, quantification of nodalization-based code uncertainty is deemed to be of lesser importance relative to the practical requirements of model accuracy and calculation efficiency or economics. The objective is to define the minimum noding needed to capture the important phenomena. The selection process used to arrive at this objective becomes the standard nodalization procedure. The standard nodalization procedure is applied to every code assessment and LBLOCA analysis; thus, minimizing nodalization as a contributor to uncertainty.

Code assessment using the test matrix from Step 7 and the nuclear power plant nodalization of Step 8 is used to accomplish Step 9, code, and experiment accuracy. Code accuracy is quantified for bias and deviations through confirmatory code uncertainty analysis and benchmarks. This step also serves as a validation for Step 6, code applicability, and sets up the tasks of element 3, sensitivity, and uncertainty analysis. The demonstration of code accuracy—or for a conservative EM, code adequacy—has always been a required component of LOCA evaluation methodologies. With a CSAU-based evaluation methodology, the emphasis is focused on evaluating the important individual contributors (i.e., phenomena) to the overall code uncertainty.

For the dominant LBLOCA phenomenon (e.g., critical flow, film boiling, condensation, fuel stored energy, etc.), sets of separate effects tests were used to derive the S-RELAP5 code uncertainty as it relates to each individual phenomenon. From the code-to-data comparisons, such as that seen in Figure 3 comparing S-RELAP5 results (x_c) to Marviken critical flow test data (x_m), code bias (μ_x) and the statistical standard deviation (σ) were evaluated.

While uncertainty quantification obviously requires data, the process for quantification begins with a clear qualitative understanding of the assumptions associated with measured values. This is the nature of probability and statistics in general. For example, heat transfer is fundamentally dependent on geometry, power,temperatures, fluid properties, and mass flow. In a nuclear power reactor core, heat transfer is complicated by multidimensional effects resulting from core and fuel design and radial and axial power variations. In addition, potentially dramatic changes in fluid properties can occur as a consequence of both phenomenological (e.g., phase change) and plant process response (e.g., safety injection). However, what we know about core heat transfer has been gathered from data taken from prototypical systems of likely different scale skewed by limitations in measurement capabilities and data reduction techniques.

The quality of the data, characterized by both quantitative limitations such as the domain of system conditions during testing and qualitative limits associated with measurement factors and data reduction, must be addressed. The ideal nature of measured data would have the following characteristics.

(i) Phenomenon of interest is measurable independent of other phenomena.
(ii) Phenomenological dependencies with a particular system condition are measurable independent of changes of other system conditions.
(iii) Detailed dimensional variations are measurable.
(iv) Scale distortion is eliminated.

Since real data often does not have these characteristics, data reduction techniques have been devised and applied to compensate. Such methods often involve the elimination of "tainted" data and/or the averaging of data. The cost of such techniques is typically seen in the loss of some data and/or the broadening of uncertainty measures. Some examples from a hypothetical reflood heat transfer test are as follows:

(a) the elimination of temperature data for heater rods near a "cold" vessel wall (possible excessive radiation)—a consequence of scale distortion;
(b) insufficient number of thermocouples to track radial and/or axial temperature variation in the simulated fuel assembly resulting inthe need to track computed average temperature results or just the peak temperature results by eliminating data not considered "peak";
(c) tracking a total heat transfer measure rather than separate heat transfer mechanisms and other influencing phenomena (i.e., combinations of radiation and convection between walls and liquid and vapor fluids, interfacial drag); that is, tracking the convolution of multiple phenomena to produce an "aggregate phenomenon";
(d) binning temperature data over a segment of a test condition range (e.g., pressure, void fraction) to assure an adequate depth of data necessary to generate meaningful uncertainty measures.

Such limitations in data are manageable; however, the implications of such limits should be addressed in the implementation of the uncertainty measures used in BEPU methodologies.

Completeness requires that the treatment of each important LBLOCA phenomena be addressed; however, a full quantification of uncertainty for each phenomenon is not necessary and, given the availability of data, may not be possible. "Phenomenological treatment" should describe a method in which the parameter range of each LBLOCA

Table 3: Summary of S-RELAP5 assessment matrix.

Test facility	Tests used	Key phenomena of interest	References
THTF heat transfer	35	Heat transfer	[25–28]
THTF level swell	3	Void distribution	[29]
GE level swell	1	Void distribution	[30]
FRIGG-2	27	Void distribution	[31]
Bennet tube	2	Heat transfer	[32]
Flecht and flech-seaset	9	Heat transfer, nodalization, axial power distribution, scalability, entrainment	[33, 34]
PDTF/SMART	4	Spacer effects	[35]
Marviken	9	Break flow	[36]
W/EPRI 1/3 scale	9	Cold leg condensation, interfacial heat transfer	[37]
MiniLoop CCFL	3	Upper tie plate CCFL	[38]
Multidimensional flow	3	Core flow distribution	[39]
UPTF	14	ECCS bypass, steam binding, CCFL, scalability, nodalization	[40–45]
CCTF	4	Steam binding, nodalization, scalability	[43–50]
SCTF	6	Nodalization	[51]
ACHILLES	1	Accumulator nitrogen discharge	[52]
LOFT	4	Overall code performance, nodalization, scalability	[53–57]
Semiscale	2	Blowdown heat transfer, nodalization, scalability, compensating errors	[58–60]

contributor is covered. The use of statistics provides various methods for describing ranges of uncertainty for a given problem; however, the CSAU process does allow for methodology conservatisms to satisfy the objective of defining uncertainty treatment for individual code models and correlations. The practical limitations of economics and data availability are considered when accepting a conservative phenomenological treatment. The trade off is the reduction in margin relative to the LBLOCA acceptance criteria. Again, engineering judgment can play a role in how to approach this step. Table 4 provides a summary of the parameters for which code uncertainty was quantified. While in most cases AREVA NP developed proprietary analyses to quantify parameter uncertainty, quantified uncertainty for a few parameters appears in open literature. In those cases (i.e., metal-water reaction and decay heat), the values used in the AREVA NP RLBLOCA methodology are provided.

Given quantified uncertainty measures, the integrity of the statistics requires the demonstration of sufficient density and breadth of data within the range-of-applicability. Validation of uncertainty ranges or standard deviation is provided by reserving "control sets" of data and reevaluating statistics. Data from integral effects tests (e.g., CCTF, LOFT, and semiscale) was used to demonstrate the acceptability of the code biases developed from the separate effects tests. Figure 4 shows a comparison of a CCTF Test 54 assessment before and after the evaluation of code biases.

Beyond the uncertainty quantification exercise, the primary challenge of Element 2 is to demonstrate sufficient range of applicability of the computer code models and correlations. Code models and correlations are best assessed using separate effects test data developed for the explicit purpose of investigating the phenomena described by the code model or correlation. Establishing a sufficient range of applicability is complicated by the fact that conditions

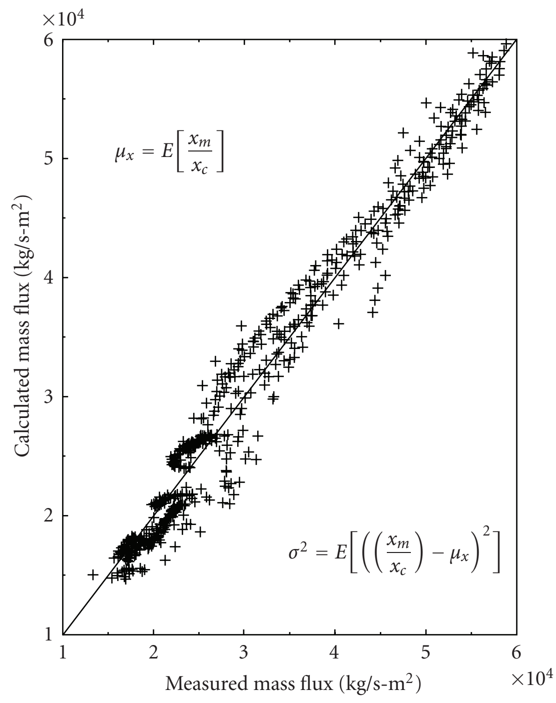

Figure 3: Calculated versus measured results for Marviken critical flow tests.

Table 4: Summary of uncertainty quantification exercise.

PIRT Parameter	Bias	σ	Min or -2σ	Max or $+2\sigma$
Break Size	N/A	N/A	0.1	2
Break discharge coefficients	###	###	###	###
Critical heat flux	###	0.0	###	###
Film boling HTC	###	Special	###	N/A
Dispersed film boiling	###	Special	###	N/A
Tmin,	###	###	###	###
Power (inc. radial and axial shapes)	Treated as a sampled plant parameter			
Stored energy (centerline temperature)	###	###	###	###
Metal-water reaction constant	1	0.182	0.636	1.364
Metal-water reaction exponent	1	0.0134	0.9732	1.0268
Decay heat uncertainty	1	0.003	0.94	1.06
Condensation interface HTC	###	Uniform	###	###
Steam generator Inlet Interphase Friction	###	0.0	###	###
Hot wall (CHF multiplier)	###	Binary	###	###
Containment pressure (volume)	###	Uniform	Min free volume	Max free volume

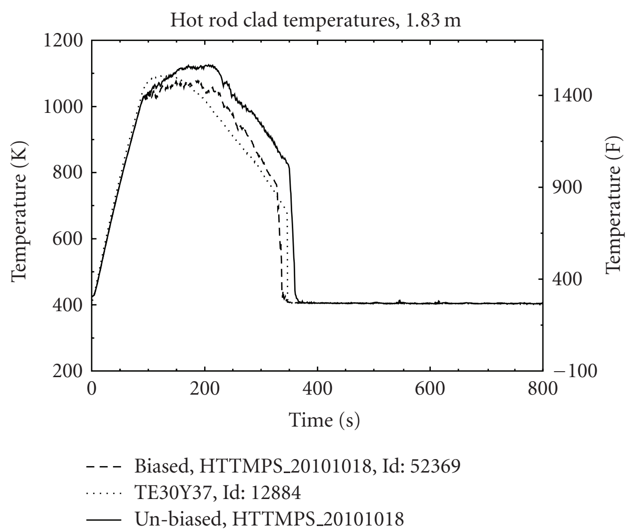

Figure 4: Comparison of CCTF Test 54 assessment before and after the evaluation of code biases.

present during a PWR LBLOCA span thermal-hydraulic ranges (pressures, temperatures, flows, etc.) that exceed the ranges of any individual separate effects test. Given this inherent limitation, the logical approach to establish the pedigree of a particular code model or correlation must incorporate a broader body of knowledge on the phenomena of interest. Applying an analogy from vector space analysis, the "applicability space" will not only include data from various separate effects test programs, but also analytical solutions and data from various integral effects tests. It is the collection of this full body of phenomenological knowledge: the analytical model, the statistical description of uncertainty from separate effects tests, and validation with integral effects tests—as incorporated within a calculational framework such as S-RELAP5 that provides the technical basis supporting the declared range of applicability of a code model or correlation.

An added complexity to the applicability question is test scalability.This is addressed in Step 10. In the long history of thermal-hydraulic code models and correlations development, computer code models and correlations have often been "tuned" to particular data sets. This approach to computer code development can create a results bias and uncertainty associated with the scaling of the problem of interest. Scaling uncertainty can be evaluated using data from a suite of test programs generated at various scales. For the specific application to the PWR LBLOCA, there is a motivation to acquire full-scale data for the dominant LBLOCA phenomena. Fortunately, many hydraulic phenomena can be assessed using tests performed at the full-scale upper plenum test facility (UPTF). In addition, heat transfer phenomena can be assessed applying data from the many reflood tests that have been performed with full-scale assemblies. The AREVA NP RLBLOCA methodology utilized the available full-scale data wherever possible. In addition, code-to-data comparisons from scaled test facilities did not show a significant scale bias. With this approach to the scaling issue, no additional accounting for scale is necessary.

3.3. *Sensitivity and uncertainty analysis*

Given the inherent uncertainty and complexity of the thermal-hydraulic processes appearing during a large-break LOCA, a best-estimate statement of assurance must be provided statistically. This CSAU element focuses on setting-up, executing, and evaluating a RLBLOCA analysis. As a statistics-based methodology, the problem setup involves implementing the bias and uncertainty for the LBLOCA contributors identified from CSAU Elements 1 and 2. Execution involves the convolution of these uncertainty contributors and the final result is evaluated from the number of calculations necessary to provide a statistically meaningful set.

While the CSAU methodology through Step 9 is focused on phenomenological contributors to uncertainty, it recognizes in Step11 that there is also uncertainty associated with the measurable states that define a plant's operating condition, such as pressures, temperatures, levels. For utility customers interested in plant-specific application of an approved methodology, this step may be the most important step; however, the CSAU methodology [4] discussion provides the least amount of direction. In response to the limited amount of guidance provided by the TPG, the AREVA NP approach has been detailed and reported in [61].

The key challenge to addressing the uncertainty associated with plant state is reconciling the requirement for analyses to support a plant's licensing basis through the plant's design and control specifications while still being "best-estimate." Traditional deterministic analyses explicitly utilize a plant's technical specifications when it is clearly conservative to do so; otherwise, a best-estimate value is considered to bound the technical specification. Since no provision is made for BE methods to exempt the use of conservative technical specification in safety analysis, the concept of "realistic conservatism" is unavoidable. That is, this condition is a function of the regulatory process for plant licensing and not an artifact of the developed safety analysis methodology.

AREVA NP's approach to identify which plant parameters to explicitly treat as an uncertainty parameter, either as a direct bias or sampled, considers the interests of several constituents. The primary regulatory interest requires that the plant be analyzed at technical specification limits. Precedence established by Appendix K methods provides the list of those parameters that are expected to be treated in this fashion. A second interest has been inferred by AREVA NP given the emphasis in the CSAU methodology on important phenomenological contributors to LOCA acceptance criteria. AREVA NP chose to recognize that plant response to an off-normal event is driven by phenomena. Specifically, plant parameters were correlated to phenomena and the importance of a plant parameter was made in relation to any associated phenomenological parameter. For example, accumulator pressure will affect ECCS bypass and initial flow rate will affect break flow. In effect, the inclusion of a plant parameter's operational and measurement uncertainty implicitly broadens the range and distribution of PIRT parameters. The third interest in this regard is the customer. In this situation, the customer may be interested in an analysis of some process or condition for which an expanded operational variance is desired, for reasons beyond the normal support of a plant's limits of operation. The uncertainty treatment for these parameters is handled just like other sampled parameters.

Table 5 presents the list of plant parameters treated in the AREVA NP RLBLOCA 3- and 4-loop sample problems and their relation to important PIRT parameters. Generally, the impact of plant parameters will be much less than PIRT parameters. Most plant parameters represent initial conditions; hence, their impact diminishes with time. Typically, limiting LBLOCA safety analyses show PCT during late reflood; hence, the impact of plant initial state is likely very small. The ECCS parameters will influence the simulation throughout the event; hence, greater importance should be given to these plant parameters.

The objective of CSAU Steps 12 and 13 is to combine the bias and uncertainty of the important individual contributors as identified in Step 9 and Step 11 through the running of a large set of plant simulations. RLBLOCA simulations using the AREVA NP methodology involves two computer codes: RODEX3A and S-RELAP5. As stated in the introduction, RODEX3A is a fuel performance code that provides fuel material property characteristics that determine a fuel pin's initial stored energy versus burnup. S-RELAP5, a derivative of RELAP5/MOD2 and the CONTEMPT codes, uses the RODEX3A results to initialize the fuel heat structure models as a part of calculating the steady-state solution that initializes the LBLOCA transient simulation. S-RELAP5 is then executed for the transient simulation of the fuel and coolant system response to the break and containment back pressure condition.

The convolution of the many LBLOCA uncertainty contributors (Tables 4 and 5) to PCT is an inherently statistical approach. The two common approaches are generally classified as either parametric or nonparametric. The response surface method, a parametric method, was the approach demonstrated in the CSAU sample problem [4]. The objective of that method is the development of a response surface describing peak clad temperature sensitivity to the dominant LBLOCA uncertainty contributors. The number of calculations required for that approach is dependent on the number of LBLOCA uncertainty contributors considered. AREVA NP chose to apply a nonparametric approach originally recommended in the German Gesellschaft fur Anlagen und Reaktorsicherheit (GRS) methodology [62]. This statistical method is often referred to as Wilks' method [63]. The nonparametric approach decouples the association between the number of uncertainty parameters and the number of required calculations. The desired quantification of PCT uncertainty is the identification of a specific result that represents coverage of the results domain at or above 95% with a 95% confidence. The 95/95 coverage/confidence has been recognized by the USNRC having sufficient conservatism for LBLOCA analyses.

The minimum number of sampled cases is given by Wilks' formula for one-sided tolerance limits. Beginning with the probability statement

$$P[F(x_k) > \beta] = \frac{n!}{(k-1)!(n-k)!}\int_{\beta}^{1} \xi^{k-1}(1-\xi)^{n-k} d\xi, \quad (1)$$

where the $P[F(x_k) > \beta]$ is the "probability that the result from a given sample case ($F(x_k)$) exceeds the β percentile" case. When $k = n$, that is, the largest value of all of the samples, this relationship reduces to

$$\gamma = 1 - \beta^n, \quad (2)$$

where β is the coverage, γ is the confidence, and n is the minimum number of sampled calculations. For the 95/95 coverage/confidence condition, $n = 59$. This means in a random

Table 5: Treated process parameters and relation to PIRT.

Process parameter	Influenced phenomenon (PIRT subset)
Fuel state (burnup and power peaking)	Stored energy
Core power	Stored energy
Power peaking, axial shape	Stored energy
Loop flow rate	Flow split, DNB
Core inlet temperature	DNB
Upper head temperature	Flow reversal, stagnation
Pressurizer pressure, level	Early quench, critical flow in surge line
Accumulator pressure, temperature, level	Accumulator discharge, condensation, noncondensable gases
Containment volume, heat transfer, sprays	Backpressure, critical flow
Steam generator feedwater temperature	Core heat transfer
Offsite power and diesel start delay	Core heat transfer via pumped ECC

sample of 59 calculations, one case, the highest PCT case, will bound the 95/95 coverage/confidence condition for PCT. A disadvantage of this method is that there may be significant conservatism as a result of bounding the 95/95 condition. Applying Somerville's generalization of Wilk's formula on nonparametric tolerance limits [64] can improve the fidelity in the final result through the performance of additional calculations.

Each calculation is setup by first sampling every LBLOCA uncertainty contributor over its derived range. A minimum of 59 calculations are performed. The PCT results from each calculation are sorted to identify the highest PCT. The highest PCT result from 59 calculations bounds the 95/95 condition.

Included in the AREVA NP RLBLOCA methodology, topical reports are sample problems demonstrating application of this methodology on both a 3- and 4-loop Westinghouse pressurized water reactor. Some results from the 3-loop sample problem were presented in [65], which culminated in a PCT of 1853°F. For this problem, more than 30 uncertainty parameters were statistically treated using a Monte Carlo sampling procedure for the creation of 59 code input file sets. Each set included four input files describing models for the fuel performance evaluation, thermal-hydraulic steady-state initialization, thermal-hydraulic transient response, and simultaneous containment response.

The final step in the CSAU process is to identify the total uncertainty. If any PCT gains or penalties were identified during the CSAU process, they are to be applied in Step 14. In addition, the total uncertainty can be quantified relative to a "best-estimate" figure-of-merit. The total uncertainty does not have meaning in relationship to regulatory acceptance criteria. As such, the importance of this measure is somewhat diminished from what the TPG originally envisioned. AREVA NP chose to define total uncertainty using the 50/50 condition, also evaluated from nonparametric statistics. The 50/50 condition is provided by the calculation providing $(n+1)/2$ for an odd-numbered sized sample space. For the sample problem, the 50/50 condition was identified as 1500°F; hence, the 95/95 condition represents about 350°F uncertainty.

4. REGULATORY REVIEW

The unwritten "Element 4" in the CSAU process is the USNRC regulatory review process. This process spanned over 20 months and required 139 formal "requests for additional information." Plant-specific elements of the generic review were addressed for the first application and an additional 12 months and approximately 30 RAIs were required. The bulk of the review focused on the explicit definition of the range of applicability for the key LBLOCA phenemological and plant parameters. This was provided following the methods previously discussed in the Element 2 section. In addition, the USNRC requested technical basis supporting the treatment of fuel relocation, downcomer boiling and rod-to-rod radiation–phenomena not appearing on the AREVA NP PIRT. AREVA NP responded to these concerns by supplying new sensitivity results and/or detailed characterization of how the existing model was adequate.

5. CONCLUSION

The AREVA NP RLBLOCA methodology is a CSAU-based methodology for performing best-estimate large-break LOCA analysis. The methodology addresses all of the expressed steps of the CSAU process. The key challenge to this process has been the defense of declared engineering judgment and the demonstration of the methodologies range of applicability. This was accomplished by careful characterization of dominant LOCA parameters and emphasis on validation through sensitivity studies and the statistical nature of the methodology.

The generic AREVA NP RLBLOCA methodology was approved by the USNRC in April 2003 and is now being applied to several nuclear power plants serviced by AREVA NP Inc. While the CSAU methodology represents a significant departure from traditional deterministic methods, the AREVA NP methodology applying nonparametric statistics retains an economical viability on par with existing methodologies. Throughout the 40+ staff-years of development effort at AREVA NP, the CSAU process has withstood the technical questions and challenges to its foundation. The key benefits realized by AREVA during this development are

(i) The move to a realistic LOCA methodology brings a new clarity of understanding of the LBLOCA problem to the industry by demonstrating contrast to the very conservative 10 CFR 50 Appendix K methodologies.

(ii) Through use of statistically-based methods, there is improved characterization of the conditions in which individual LBLOCA uncertainty contributors influence LBLOCA response.

(iii) The reliance on experimental data has revived the importance of the many test programs that have long since been decommissioned.

These rewards alone have validated the CSAU approach.

ACRONYMS

CSAU: Code scaling, applicability, and uncertainty
ECCS: Emergency core cooling system
GRS: Gesellschaft fur Anlagen und Reaktorsicherheit
HEM: Homogeneous equilibrium model
LOCA: Loss of coolant accident
PCT: Peak clad temperature
PIRT: Phenomena identification and ranking table
PWR: Pressurized water reactor
RLBLOCA: Realistic large-break LOCA
TPG: Technical Program Group
USNRC: United States Nuclear Regulatory Commission

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Research Article

A Statistical Methodology for Determination of Safety Systems Actuation Setpoints Based on Extreme Value Statistics

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This paper provides a novel and robust methodology for determination of nuclear reactor trip setpoints which accounts for uncertainties in input parameters and models, as well as accounting for the variations in operating states that periodically occur. Further it demonstrates that in performing best estimate and uncertainty calculations, it is critical to consider the impact of all fuel channels and instrumentation in the integration of these uncertainties in setpoint determination. This methodology is based on the concept of a true trip setpoint, which is the reactor setpoint that would be required in an ideal situation where all key inputs and plant responses were known, such that during the accident sequence a reactor shutdown will occur which just prevents the acceptance criteria from being exceeded. Since this true value cannot be established, the uncertainties in plant simulations and plant measurements as well as operational variations which lead to time changes in the true value of initial conditions must be considered. This paper presents the general concept used to determine the actuation setpoints considering the uncertainties and changes in initial conditions, and allowing for safety systems instrumentation redundancy. The results demonstrate unique statistical behavior with respect to both fuel and instrumentation uncertainties which has not previously been investigated.

1. INTRODUCTION

In existing and new nuclear power plants, a variety of special safety systems are employed which will trigger fast reactor shutdown in the event of an accident or undesirable plant condition. These special safety systems utilize multiple and redundant measurements of certain process and neutronic variables, known as trip parameters, which are continuously monitored against predetermined limits. If a measured trip parameter deviates in an unsafe direction in excess of these predetermined limits, known as trip setpoints, the special safety system will initiate a fast reactor shutdown. Nuclear safety analysis is performed to determine the plant response to hypothetical accident scenarios and to assess the effectiveness of the trip parameters and setpoints in achieving the safety goals (i.e., precluding fuel failures or minimizing public dose). Hence, nuclear safety analysis is a critical component in the operation and regulatory licensing of nuclear power plants.

Historically, a set of bounding analysis methodologies and assumptions were used to determine plant response to these events. As a result of these simplifications, it is impossible to determine the exact margins to safety limits. Furthermore, due to scientific discovery issues combined with plant safety margin deterioration due to component aging, these traditional methodologies predict consequences which may prohibit full power operation. In addition to the above, changes in the regulatory framework for operating reactors are also driving changes in the methodology used to demonstrate plant safety [1]. Furthermore, risk-informed decision (RID) making practices and maintenance optimization [2] at each plant rely on accurate quantification of the impact of upgrades/refurbishment on safety margins. The Canadian Nuclear Safety Commission (CNSC) and the USNRC have recognized that best-estimate predictions of plant response, along with accurate assessments of uncertainties, are an acceptable alternative to more limiting and bounding analyses for demonstrating safety system response [3, 4].

The Canadian CANDU industry is currently pursuing the use of best-estimate and uncertainty (BEAU) methodologies to resolve various issues related to loss-of-power regulation, loss-of-coolant and loss-of-station power accidents [5]. Due to computational limitations, the most recent efforts within the CANDU industry have utilized best-estimate simulations of the liming fuel channel or detector system within the core. Extensions of best-estimate methodologies to include the effects of the minimization and maximization over the entire core of fuel channels in a CANDU have been performed by Sermer et al. [6, 7], to examine the uncertainty in predicting the maximum fuel-channel power, and by Pandey [8], pressure tube integrity issues. Furthermore, the applications of extreme-value theory are also important in the finance and insurance industries [9] as it can provide estimates of both the likelihood and confidence of rarely occurring events.

The use of extreme-value statistics provides a more accurate framework for establishing the uncertainty in the estimated outcomes by examining not just the uncertainty in individual fuel channels or trip instrumentation responses, but rather the uncertainty in computing maxima and minima of the quantity in question. This paper presents a methodology for determining the required trip setpoints during transient accident analyses of special safety systems using the so-called extreme-value statistics and accounting for the multiple and redundant measurements available within each safety system.

2. BACKGROUND

For a typical CANDU reactor, there are 480 fuel channel assemblies in the reactor core which are fed by two separate figure-of-eight heat transport system loops. Each figure-of-eight loop has 2 heat transport system pumps and 2 steam generators for heat removal and provides coolant flow to half of the fuel channels. The 480 fuel channels contain from 12 to 13 natural uranium fuel bundles at power levels up to approximately 6 mW per channel. A heavy water moderator surrounds each fuel channel assembly and is contained in a calandria vessel. Reactor power is controlled through the reactor regulating system (RRS) which manages bulk and local power levels, as well as monitoring of the core for abnormal occurrences. In the event of abnormal operating occurrences or accidents, regulatory requirements are placed such that fuel and pressure tube failures are precluded. Defense-in-depth was typically employed such that there is a large margin to fuel and pressure tube failure at the time of safety system actuation.

CANDU reactor designs operate at much lower heat fluxes than light water reactor (LWR) designs, and hence the use of dryout (or in the LWR case, departure from nucleate boiling) as an acceptance criteria is excessively conservative since the sheath and fuel temperature excursions in the postdryout regime are much more benign than that under similar LWR conditions. Therefore, for actual CANDU applications, it has been recommended that alternative thermalhydraulic criteria, such as prevention of sheath temperatures exceeding 600°C, be adopted. However to simplify this methodology, and for consistency to common LWR acceptance criteria, the acceptance criteria adopted in this paper will be the prevention of dryout in all fuel channels

CANDU reactors are equipped with two independent shutdown systems, each with the capability of rendering the core subcritical and each with its own unique set of instrumentation. The instrumentation systems within each shutdown system are divided into three logic channels and within each logic channel there are several redundant instruments measuring plant variables. The shutoff mechanism relays are actuated when trip signals from two-out-of-three exceed their trip setpoint. In the event of an accident at a CANDU station, the transients may be terminated by the RRS monitoring systems or either of the special safety shutdown systems.

Nuclear safety analyses are performed for selected accident scenarios to determine both the setpoints required for shutdown system instrumentation and accident consequences. Computer codes are used to model reactor core physics and heat transport system behavior during postulated transients; and the code predictions are used to establish the trip setpoints required to prevent undesirable consequences. The original nuclear safety analysis for CANDU stations was performed using deterministic assumptions such that the consequences demonstrated in the analysis bounded all possible outcomes for that accident scenario and to provide the most conservative estimate of the required actuation setpoints for the special safety systems. In order to better estimate the actual margins, to provide input for risk-informed decision making, and to better focus plant upgrade activities, best-estimate safety analyses are being proposed as part of the continuous nuclear safety analysis update program. With the advent of statistical methodologies, the focus has now shifted to providing shutdown system trip setpoints with very high probability, or alternatively assessing the probability of failure with existing setpoints. This paper presents the framework for this methodology and demonstrates the application to a simplified bulk power excursion event.

3. METHODOLOGY

3.1. Required trip setpoint

The methodology proposed in this paper provides a statistical treatment of the available instrumentation response as well as the fuel-cooling response which may be applied to best-estimate analyses. Consider a certain accident scenario in a nuclear power plant at a fixed instant in time. For this scenario, there is some value of the shutdown system activation trip setpoint, *tsp*, which will initiate shutdown such that the safety objectives are met. The value of this trip setpoint could be determined if

(i) the initial operating conditions at that instant were known exactly,

(ii) the simulation of the plant response was without error, and if

(iii) the actual safety system measurements were perfect.

Given the above, a setpoint for each shutdown parameter could then be determined based upon the value of the key instrumented physical at their specified locations in the reactor. This true trip setpoint would provide 100% probability that the safety objective would be met if an accident occurred at that instant in time. In reality, the true setpoints cannot be known due to uncertainty in the models used to predict the outcome and uncertainty in the initial conditions at that instant in time. Even if the true trip setpoint could be established at a given instant in time, the acceptance criterion may still be violated due to uncertainty associated with each instrument used in the special safety systems. Finally, since there are variations in the actual plant conditions caused by fuel burn-up, process system variability, and plant-component aging, these must also be considered in setpoint determination.

What is needed is a required trip setpoint (RTSP) which will cause a reactor shutdown such that there is high probability that the acceptance criteria will be met at a certain reactor configuration, m. The RTSP should account for: (i) the uncertainty in instantaneous plant boundary conditions, (ii) the uncertainty in simulation models and computer codes used to predict the plant response, (iii) the measurement uncertainties related to shutdown system instrumentation, and (iv) the instrument time delays and uncertainties in time delay if necessary. (It is assumed that the instrument response and reactor shutdown on a trip signal are prompt with respect to any true value change. These assumptions are not necessary for this methodology, but are made to simplify the following calculations. Modified derivations are available to account for instrument and shutdown response characteristics.) Once the RTSP for state m is established, a large number of reactor states could be examined and an appropriate statistical lower bound could be determined based on the RTSP for each $m+1$ considered. The application of the methodology for time-dependent reactor states is discussed in the subsequent sections.

The true trip setpoint for an instantaneous reactor state, tsp_m, is defined as the setpoint required to meet the acceptance criterion given complete knowledge of the initial plant conditions at that instant, perfect computational models for that accident sequence, and perfect measurements. Since these conditions, models, and measurements are not perfect, only an estimate of the setpoint, TSP_m is available. The relationship between this estimate and true value is given as

$$\mathrm{TSP}_m = \mathrm{tsp}_m(1+\varepsilon_m), \tag{1}$$

where ε_m is the error in the estimated setpoint at that instant in time and is a random variable which considers errors in the initial conditions, plant response models and instrumentation uncertainty and consequently TSP_m is a random variable. What is needed is the required trip setpoint based on the random TSP_m, which will have a high probability of

$$\mathrm{RTSP}_n \begin{cases} \leq \mathrm{tsp}_m & \text{high going limit,} \\ \geq \mathrm{tsp}_m & \text{low going limit.} \end{cases} \tag{2}$$

For simplicity, the remainder of this section will deal with the trip setpoint at a given instant in time and hence the subscript, m, is dropped. For the sake of convenience, the foregoing paper will examine high-going trip setpoint limits (i.e., a variable that will trip the reactor if it exceeds some maximum value). The application of the methodology for time-dependent reactor states is discussed in the subsequent sections; and for low-going trip setpoints, the methodology is a simple extension.

3.2. *Acceptance criteria*

As discussed in Section 2, dryout must be prevented in each of the 480 fuel channels such that

$$\min_{i=1,480}(\mathrm{mtd}_i) > 1.0 \tag{3}$$

which specifies that the minimum margin to dryout (mmtd) over the entire CANDU core must be greater than unity. (For LWRs an alternative such as $(\mathrm{mtd}+\gamma)$ may be used, where γ is a predefined margin to the departure from nucleate boiling.)

Specifically, mtd_i is the true value of the margin to dryout in channel i computed from

$$\mathrm{mmtd} = \min_{i=1,480}\left[\mathrm{mtd}_i\right] = \min_{i=1,480}\left[\frac{\mathrm{ccp}_i}{\mathrm{cp}_i}\right], \tag{4}$$

where cp_i is the instantaneous channel power in channel i and ccp_i is defined as the critical channel power in channel i. The critical channel power (CCP) corresponds to the channel power that would be required to initiate dryout for the same thermalhydraulic inlet boundary conditions. During the progression of the accident, the margin to dryout will be a function of time t, and hence it is required that the minimum margin to dryout, mmtd, is

$$\mathrm{mmtd} > 1.0 \tag{5}$$

for all times of interest. Equation (5) can be reformatted using order statistics as

$$\mathrm{mmtd} = \mathrm{mtd}_{(1)} > 1.0, \tag{6}$$

where the subscript (5) indicates the smallest value in the ordered set mtd.

3.3. *Safety system actuation*

Safety and shutdown systems in a CANDU plant are actuated when the multiple and redundant special safety system instruments exceeds the trip setpoint for that variable. For the following analysis, the instrumentation response is measured as a fractional value of the trip setpoint and denoted as f_j, where j is the instrument number. Furthermore, the analysis will consider one shutdown system with instruments grouped into one of the three logic channels labeled D, E, and F. Within each logic channel, instrumentation measures the plant response and compares the measured value to the predetermined trip setpoint; and if it exceeds this threshold, a trip will register on that logic channel. As mentioned, if two-out-of-three logic channels register a trip, the safety system will activate.

At the point in the accident transient where the margin to dryout approaches unity, the setpoint is selected such that at least one of the following holds:

$$\begin{aligned} &\frac{1.0}{\min\left(\max\left[f_j^{\mathrm{D}}\right],\max\left[f_j^{\mathrm{E}}\right]\right)} < 1.0,\\ \text{or}\ &\frac{1.0}{\min\left(\max\left[f_j^{\mathrm{D}}\right],\max\left[f_j^{\mathrm{F}}\right]\right)} < 1.0,\\ \text{or}\ &\frac{1.0}{\min\left(\max\left[f_j^{\mathrm{E}}\right],\max\left[f_j^{\mathrm{F}}\right]\right)} < 1.0, \end{aligned} \tag{7}$$

where D, E, and F are the labels for each of the logic channels in a safety system. The above expression ensures that in the event the margin to dryout decreases to its acceptance criteria, than the trip will actuate the shutdown system based upon 2-out-of-3 logic channels exceeding the setpoint. For comparison to order statistic approaches, the trip signals can be grouped into a single set, **s**, and the appropriate order statistic selected. Therefore, **s** is given as

$$s = \left[f_{(n)}^{\mathrm{D}}, f_{(n)}^{\mathrm{E}}, f_{(n)}^{\mathrm{F}}\right], \tag{8}$$

where the subscript (n) denotes the highest detector reading in each ordered set of responses within that logic channel. For example, for the 2-out-of-3 logic trip,

$$\mathrm{mtt} = s_{(2)} < 1.0, \tag{9}$$

where mtt is the margin to trip and $s_{(2)}$ denotes the second smallest value in the ordered set **s**. It should be noted that in many licensing applications, the goal is to demonstrate a reactor trip in the analysis on 3-out-of-3 logic channels, in which case the minimum margin to trip, mmtt, is

$$\mathrm{mmtt} = s_{(1)} < 1.0. \tag{10}$$

It can be shown that for the more general case for k-out-of-n trip logic, the proper order statistic for the margin to trip is

$$\mathrm{mmtt} = s_{(n-k+1)} < 1.0. \tag{11}$$

Hence the true trip setpoint can be selected for a given accident such that (10) holds at the point in the transient where the margin to dryout approaches unity.

3.4. Margin to dryout uncertainty

The methodology used to select the setpoint above is applicable to only situations where perfect information is available (i.e., where the true values can be established). In reality each of the variables discussed above is subjected to both measurement and simulation uncertainties which may have components that are a function of space and time. For example, instruments in different parts of the core may have differing uncertainties, the simulated transient code predictions at the measurement locations may be delayed/accelerated in time, and the critical channel power in any of the 480 channels may be over or under predicted at any instant. In addition, there may be a noise component in the actual instrument behavior.

First, consider three hypothetical CANDU reactor cores with 1 fuel channel, 5 identical fuel channels, and 10 identical fuel channels, respectively; and assume initially that there is an independent random uncertainty in the margin to dryout prediction in each channel such that

$$\mathrm{MTD}_i = \mathrm{mtd}_i\left(1 + \varepsilon_i^{\mathrm{mtd}}\right), \tag{12}$$

where $\varepsilon_i^{\mathrm{mtd}}$ denotes the error in channel i. For demonstration purposes, it will also be assumed that the errors are normally distributed, independent, with mean 0.0, and standard deviation of 4.0% (i.e., a typical value of CCP uncertainty in CANDU applications) and that the true value are equal. The estimate of the minimum margin to dryout will therefore be

$$\mathrm{MMTD} = \min_{i=1,z}\left[\mathrm{mtd}_i\left(1 + \varepsilon_i^{\mathrm{mtd}}\right)\right], \tag{13}$$

where z is the number of channels in the hypothetical reactor being considered. At a given point in an event sequence assume that the true minimum margin to dryout decreases to a value of 1.08. Monte-Carlo simulation can be performed to determine the probability of predicting a trip

$$P\{\mathrm{MMTD} \leq 1.0\}. \tag{14}$$

For the cases being considered, the probabilities are 3.2%, 9.8%, and 27.8% for the 1, 3, and 10 fuel channel reactor configurations, respectively, (the results for this simplified case of equal true values are comparable to the results obtained using the usual order statistics). This is a critical finding because it indicates that as the number of channels being simulated is increased, there is an increasing probability of declaring a false-positive when testing for fuel channel dryout (i.e., there is a 27.8% probability for a predicted value to indicate dryout when in fact the true margins were 1.08). This is to be expected because the mean of an extreme value distribution shifts in the direction of the extreme function. If at a certain point later in the transient the true margin to dryout in each channel becomes 1.01, then the probability of the estimates predicting dryout are 40.1%, 78.7%, and 99.4% for hypothetical cores containing 1, 3, and 10 fuel channels, respectively. For this simplified demonstration, it has been shown that increasing the number of fuel channels considered within the minimization process tends to increase the probability of estimating that dryout has occurred.

As an extension to this demonstration, consider the same transient but for a case where the true minimum margin to dryout has reached unity. At this point in the transient, the probability of demonstrating a trip is 50.0%, 87.6%, and 99.9%, respectively, or alternatively, there is a 50.0%, 12.4%, and 0.1% probability that dryout will not be predicted when in fact the true margin to dryout has reached 1.0 (i.e., a Type 1 error). It is clear that in considering the random nature of the several channel responses, the probability of Type 1 errors is reduced.

As an extension to the hypothetical reactor cases studies above, assume that the true values for each of the fuel

Table 1: Influence of the number of participating fuel channels in the probability of missing dryout.

q [fraction]	Number of fuel channels	Probability of predicting dryout [%]	Probability of Type 1 error [%]
1.08	1	4.7	0.0
	2	9.3	0.0
	3	13.7	0.0
	5	21.5	0.0
	10	38.4	0.0
1.04	1	19.3	1.1
	2	35.8	2.0
	3	47.8	2.1
	5	66.4	2.4
	10	88.6	1.5
1.02	1	34.0	7.0
	2	55.4	8.1
	3	70.6	8.1
	5	86.2	5.2
	10	98.3	1.1
1.00	1	50.3	18.7
	2	75.4	14.3
	3	88.2	9.1
	5	96.9	2.6
	10	99.9	0.1

channels are not equal. For this demonstration, a set of random true values is selected for each channel based on a normal probability distribution of ±2% (typical scatter in margin to dryout in a CANDU reactor for the high-power channel) centered about a mean value of q. For this set of true values, Monte-Carlo simulations were performed with random, normal, and independent uncertainties assigned to each channel. The probability of predicting dryout was recorded along with the probability of a Type 1 error given as

$$P\{\text{MMTD} > 1.0 \mid \text{mmtd} \leq 1.0\}. \tag{15}$$

The process of generating an initial set of true margins, then performing Monte-Carlo simulations about these values, was repeated a large number of times to determine the average probability of predicting dryout along with the average probability of creating a Type 1 error (the total number of simulations exceeded 10^6). The results of this study with no additional allowances are shown in Table 1.

The above example is for the special case where all fuel channels have margin to dryout within 2% and where the uncertainty in estimation is 4%. Table 1 shows that as the mean of the true margin to dryout decreases, the probability of predicting a trip increases for a core with a fixed number of fuel channels. Further, it shows that for a fixed mean true value, the probability of predicting a trip increases with the number of channels. The Table also shows that the probability of a false-negative, that is, predicting no dryout when indeed it has occurred, behaves nonmonotonically with respect to the number of channels considered or a typical Type 1 statistical error. The fundamental behavior that leads to this nonmonotonic nature has to do with the minimization function being performed. For example, in each permutation of true values for the simplified 2 fuel channel core there is a certain probability that channel A will have to lowest true margin to dryout. However, when the Monte-Carlo uncertainty simulation is performed considering the errors in estimating the margin to dryout, there is a nonzero probability that the predicted value in channel B will be lower than the predicted value of channel A. Therefore, for permutations where the estimate in channel A is in an unsafe direction, there is a probability that the estimate in channel B will be such that it compensates for that error. Note for this situation, the channel with the lowest margin to dryout was incorrectly identified, but the error in channel B assists in reducing the probability of an overall false-negative prediction in the absolute minimum over channel A and B. The larger the number of channels considered, the larger the potential for a prediction to compensate for a nonconservative prediction in channel A.

Figure 1 shows the probability of missing a real occurrence of dryout as a function of the reducing initial true margin to dryout in the channels for results considering 1, 2, 3, 5, and 10 fuel channels. As the value of the mean margin to dryout in the figure decreases, there is an increasing probability that dryout may physically occur in one or more channels. As the margin decreases to 1.0, it is evident from the figure that for estimates involving small numbers of, or single, channels the probability of missing dryout increases significantly. This is contrary to the nonmonotonic nature of the cases involving 5 or more fuel channel estimates, where the probability of missing dryout reaches a maximum and then decreases. For the hypothetical case considered when 10 or more fuel channels have true values within a band of 2%, there is less than a 2% probability of missing over the entire range of possible margins to dryout. This is a significant conclusion as it indicates that the best estimate of the minimum margin to dryout over the 10 channels provides a very accurate indication of actual occurrences of dryout.

Within the CANDU nuclear industry, this type of behavior is commonly termed extreme value statistics (EVS) since the behavior results from maxima and minima functions as applied to the random variables of interest [7]. This has extremely important ramifications in the level of probability assigned to dryout in probabilistic methods, and indicates that traditional best estimate CANDU approaches which utilize best estimate simulations for the limiting channel response are inappropriate. For any best-estimate analysis, all fuel channels, or alternatively the group of channels where the minimum margin to dryout may occur, must be considered in order to capture the true probabilities related to accident consequences. Fuel channels that have a nonzero probability of containing fuel that may undergo dryout are often termed *participants*. This terminology reflects the fact that these specific channels have a reasonable statistical probability of participating in the maximization or minimization functions.

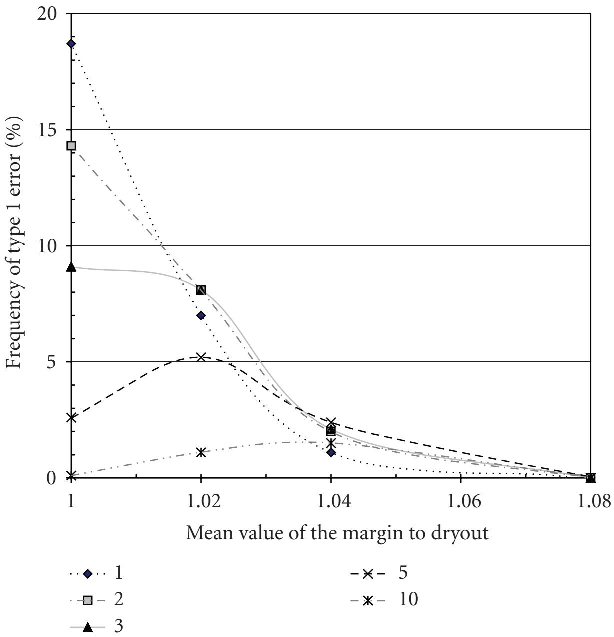

Figure 1: Probability of not predicting dryout when dryout has actually occurred for hypothetical cores with 1, 2, 3, 5, and 10 fuel channels.

It is clear that in the application of the parental errors to the margin to dryout, not all components will behave in an independent manner. For example, for fuel channels connected to common reactor inlet headers in a CANDU reactor, a component of the flow, temperature, and pressure uncertainties which lead to CCP uncertainties may be common to all channels in that core pass (i.e., an uncertainty in a header system response based on computer code such as CATHENA or TRACE will cause a common uncertainty in the margin to dryout in all fuel channels connected to that header). Therefore, an error structure is required of nature:

$$\mathrm{MTD}_i = \mathrm{mtd}_i(1+\varepsilon_i^{\mathrm{mtd}})(1+\varepsilon_{\mathrm{common}}^{\mathrm{mtd}}), \tag{16}$$

where $\varepsilon_{\mathrm{common}}$ represents a common error associated with a group of channels in the core; and ε_i is the channel specific component of the error.

3.5. Instrumentation response uncertainty

For the special safety system, instruments estimates of the results will deviate from the true values due to

(i) computer code simulation uncertainties, and

(ii) errors in the simulation of the time response characteristics of the measurement device.

Hence for each instrument, the simulated response, F_j, will be

$$F_j = f_j(1+\varepsilon_j^f), \tag{17}$$

where ε^f is the error in simulation of the instrument response. For a high going limit, the instrument with the largest response in each logic channel will initiate a trip of that channel. Therefore, for a 3-out-of-3 trip requirement, the estimated minimum margin to trip at each instant in the transient is given as

$$\mathrm{MMTT} = \frac{1.0}{S_{(1)}}, \tag{18}$$

where **S** is defined as

$$S = [F_{(n)}^{\mathrm{D}}, F_{(n)}^{\mathrm{E}}, F_{(n)}^{\mathrm{F}}], \tag{19}$$

and (n) denotes the highest reading in each ordered set of **F**. Alternatively, the minimum margin to trip error can be defined using

$$\mathrm{MMTT} = \mathrm{mtt}(1+\varepsilon^{\mathrm{mmtt}}), \tag{20}$$

where $\varepsilon^{\mathrm{mmtt}}$ is the error in the minimum margin to trip and is a complex function of the number of instruments in each logic channel and the simulation uncertainty in each instrument.

Similar to the exercise performed on the margin to dryout, an exercise is provided to illustrate these concepts for the margin to trip variable. For this demonstration, various amounts of instrument redundancy in each logic channel are considered (from one instrument per channel up to 4 responding instruments per channel) and 3-out-of-3 trip logic is assumed. A set of true values is randomly generated for each instrument about a mean value as shown in Table 2 and with a standard deviation of 3%. For a given set of true values, a Monte-Carlo analysis is performed by applying a random, normal, and independent uncertainty with standard deviation of 3% to each detector and then computing the simulated minimum margin to trip as shown in (22). The probability of simulating a safe margin to dryout for cases where the true margin falls below unity is then determined from

$$P\{\mathrm{MMTD} > 1.0 \mid \mathrm{mmtd} \leq 1.0\}. \tag{21}$$

This entire process is then repeated a large number of times for a new set of randomly selected true instrument responses and an average is then determined. The results of this exercise are shown in Table 2.

Based on these results, the probability of predicting a trip increases with the number of detectors as expected since there is a larger probability that at least one instrument will read sufficiently high to actuate the logic channel for any random perturbations. The probability of predicting a reactor trip increases as the mean of the true instrument response approaches the trip setpoint as expected. This is expected as the maximization will tend to increase the predicted value within each logic channel. Examining the Type 1 error results shows nonmonotonic behavior which is dependent on the proximity of the true instrument responses to the trip setpoint and the number of instruments within each logic channel. This Table shows a fundamental difference in the

Table 2: Influence of the number of available detectors on the probability of missing a required trip.

Mean true detector reading	Instruments per Logic channel	Probability of trip [%]	Probability of Type 1 error [%]
0.90	1	0.0	0.0
	2	0.0	0.0
	4	0.0	0.0
0.95	1	0.1	0.2
	2	1.0	0.9
	4	4.7	2.0
0.98	1	2.8	4.4
	2	15.2	12.9
	4	45.4	17.7
0.99	1	6.2	10.3
	2	27.2	23.2
	4	66.7	18.3
1.00	1	12.4	20.0
	2	42.2	25.6
	4	83.0	12.4

behavior of the trip instrumentation system as compared to the fuel channel dryout cases described previously. Although increasing the number of instruments may improve the availability of the logic system for the purposes of reliability assessments, it has a negative effect in terms of the trip predictive capability. Specifically, if a single instrument is overpredicted within the logic channel, it will cause the logic channel to trip erroneously; and, hence, the more instruments within each of the logic channels, the more probable that a single prediction will occur which trips that logic channel; when in fact the true values would indicate otherwise. Therefore, it is crucial for safety analysis predictions to include not just a single *worst responding* instrument in each channel, but rather the entire system must be simulated and the appropriate allowance or factor of safety applied.

3.6. Setpoint confidence level

Most statistical definitions for statistical setpoint and setpoint analyses, such as ISA 67.04 and CNSC regulatory guide G-144, require trip setpoints and instrumentation to provide a 95% probability with 95% confidence, or the so-called 95/95 approach. Within the context of the ISA guide [10, 11], the definition utilized for this paper is as follows:

The setpoint must provide at least a 95% probability of reactor shutdown system initiation before the acceptance criterion is exceeded with at least a 95th percentile confidence bound on the plausible reactor operating states where the setpoint need be effective.

Within the context of CANDU reactor operations, the processes show some variability such that the initial core configuration prior to an accident may take on a variety of values. Therefore, within setpoint analyses, it must be demonstrated that there is at least a 95% probability of trip over 95% of the available operating states. Practically, this can be achieved by performing uncertainty analyses about each initial reactor configurations and determining a trip setpoint that provides 95% probability of trip before the acceptance criteria, and then repeating this analysis over a large number of possible core configurations. The 95th percentile lower confidence bound over these setpoints provides will meet the 95/95 criteria specified above.

The preceding sections have examined the margin to dryout and margin to trip behavior in isolation. The following sections will integrate these results into a more realistic trip setpoint demonstration.

4. TRIP SETPOINT CALCUALTION

4.1. Trip setpoint formulation

From a given reactor initial state, it must be shown that during an accident, the margin to trip is less than one at the instant that the margin to dryout reaches unity. If the true value of all quantities were known then the trip setpoint selected would be equal to the instrument reading at the time when the true margin to dryout reached unity. The setpoint can be defined by examining an accident transient from time zero and determining the trip setpoint from the following condition:

$$\text{if}\,(\text{mmtd} \leq 1.0) \quad \text{then} \quad \left(\text{tsp} = s_{(k-n+1)}\right) \tag{22}$$

for k-out-of-n trip logic. However, due to uncertainties in the minimum margin to dryout and minimum margin to trip, detailed statistical analyses are required to assure that the required trip setpoint will actuate the reactor prior to dryout with high probability. Since the true values for each quantity above cannot be established, only the estimated trip setpoint, TSP, can be established:

$$\text{if}\,(\text{MMTD} \leq 1.0) \quad \text{then} \quad \left(\text{TSP} = S_{(k-n+1)}\right). \tag{23}$$

As stated previously, the error in this estimated trip setpoint can be established as

$$\varepsilon = \frac{\text{TSP} - \text{tsp}}{\text{tsp}}, \tag{24}$$

where ε is the error in the estimated trip setpoint. It should be noted that the error in the trip setpoint cannot be evaluated directly since it requires knowledge of the true trip setpoint. To estimate this distribution the statistical surrogate principle, or similar bootstrap method, must be employed [12]. Finally, what is required in practice is a suitable factor, η_α, which can be applied to any estimate of the trip setpoint such that the required trip setpoint meets the established probability and confidence limits for the safety acceptance criterion, that is,

$$\text{RTSP} = \text{TSP}(1 - \eta_\alpha), \tag{25}$$

where TSP is an estimate of the trip setpoint and RTSP is the required trip setpoint to ensure the safety acceptance criterion, are established to the mandated probability and confidence level. As mentioned in Section 3.6, this is determined by computing the 95th percentile error in the setpoint estimates for a large number of operating states, and taking the lower bound 95th percentile confidence level over these potential operating configurations.

4.2. Numerical demonstration

As an illustration of the setpoint methodology, consider a hypothetical bulk power excursion accident in a CANDU reactor where the true power is increasing exponentially with time constant 60 seconds and with a typical initial margin to dryout of 1.40. The assumed quantities for this case are as follows.

(i) In a given CANDU reactor, there are approximately from 10 to 20 fuel channels with very comparable margins to dryout, so that for this example 10 fuel channels are included with random initial margins to dryout characterized by a uniform distribution with mean 1.40 $\pm 3\%$.

(ii) There are typically at least 3 neutronic detectors in each logic channel which will respond to a power event so that 3 are included in this exercise along with initial detector reading with a scatter represented by a uniform distribution with $\pm 2.5\%$. Since the neutrons detectors in a CANDU are normalized to 100% FP readings and are calibrated within this band regularly, the assumed true initial detector readings have a mean of 1.0 with a uniform scatter of $\pm 2.5\%$.

Similar to the procedure in previous sections, the hypothetical true values were first randomly selected for the 10 fuel channels and the 3 detectors in each logic channel, with each of these randomizations corresponding to different possible initial reactor configurations. Then the transient was superimposed on these readings such that for this hypothetical reactor core both the true margin to dryout and true detector responses were known. Based on these transient responses, the true value of the setpoint, tsp_m, could be determined using (22). This process was then repeated by generating a new set of initial margins to dryout and trip for the channels and detectors in the core and the true trip setpoint for each core state was logged.

Monte-Carlo uncertainty calculations were then performed about each of 5000 core state utilizing the following uncertainties in key parameters:

(i) a fuel channel independent uncertainty in estimating the margin to dryout was applied to each fuel channel which was characterized by a normal distribution with standard deviation of 4%,

(ii) a random uncertainty in determining the initial margin to dryout that is common to all fuel channels and characterized by a normal distribution with standard deviation of 1% was applied. These types of uncertainties may arise from uncertainties related to common input (e.g., header inlet temperature uncertainties in a CANDU design),

(iii) a random, and detector independent uncertainty in determining the initial detector readings, characterized by a normal distribution with a standard deviation of 2%, was applied. This may be caused by uncertainties in the local reactivity during the transient or in modeling of each unique detectors neutron flux.

(iv) an uncertainty in the instantaneous power which commonly affects the margin to trip and detector readings was implemented by applying a normal distribution with standard deviation of 0.5%. This type of uncertainty is commonly associated with uncertainties related to total reactor power and/or reactivity insertion.

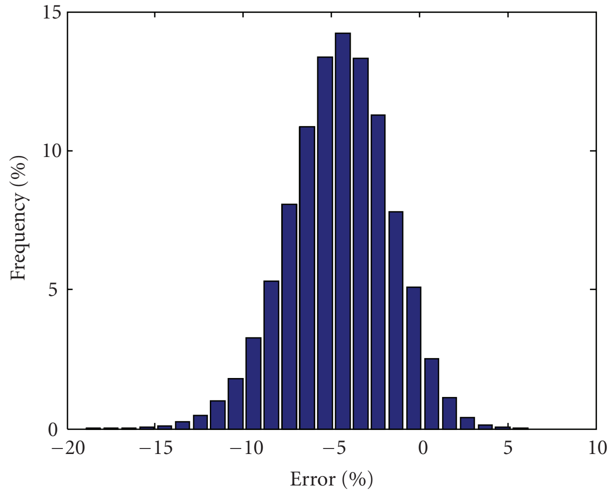

Figure 2: Trip setpoint error distribution for a selected core state.

In order to demonstrate the statistical methodology, the Monte-Carlo procedure was implemented as follows:

(i) an initial core state, m, was selected from the 5000 cases and the transient power applied to each variable. For the selected core state, the true value of the trip setpoint was determined using (22).

(ii) for the selected core state a set of estimated variables, m, is generated for each channel and detector using the uncertainty distributions outlined above. The transient power was then applied to these values along with the uncertainty in instantaneous power by using discretized time steps on the order of 0.05 second.

(iii) based on the transient behavior of the estimated variables, an estimated setpoint was determined using (23).

(iv) an error was then calculated as the difference between the estimated and true setpoints using (24).

(v) many sets of estimated variables, n, are generated (i.e., more than 1×10^5) for the hypothetical set of true values, m. The setpoints are determined and a distribution of possible errors is produced. From this distribution, the 95th percentile bounding error value can be determined. Figure 2 shows a sample of the error distribution about a selected operating state. The 95th percentile probability of the error, ε_{95}, for this initial core state was -0.004%.

(vi) a new core state is then selected, $m+1$, (i.e., a new set of true values) and the procedure outlined in steps from (ii) to (v) is repeated, and the 95th percentile error, ε_{95}, is recorded for each iteration.

(vii) A probability distribution of all ε_{95} is shown in Figure 3 based on the results of approximately 5×10^8 simulations (i.e., $m \times n$), and from this distribution an upper confidence limit on the error over all reactor states, η_{95}, is selected.

Figure 3 shows the distribution of 95th percentile errors determined based on Monte-Carlo analyses about each of the 5000 cases (i.e., based on the error determined for each of the 5000 initial core states with 1.0×10^4 Monte-Carlo passes for each state, or more than 10^7 simulations). It should be noted that the distribution is much tighter than the individual error distributions about any given single initial core state and follow a general Gumbel-type of distribution associated with extreme value statistics. The 95thpercentile upper confidence limit over all 5000 operating states considered is 1.2%, or alternatively for a 95/95 required trip setpoint the best estimate for a given reactor configuration would need to be reduced by 1.2%.

This 95th percentile confidence limit over all of the 95% probabilities for each core state provides a 95/95 probability and confidence statement which is consistent with that defined in ISA 67.04 for safety instrumentation requirements. Finally, the value of η_{95} can be used to determine the required trip setpoint based on an estimated trip setpoint using

$$\mathrm{RTSP} = \mathrm{TSP}(1 - \eta_{95}). \tag{26}$$

Equation (26) utilizes the statistic η_{95} to modify the best estimate trip setpoint, TSP, such that RTSP will provide a trip prior to dryout with high confidence. Note that depending on the number of fuel channels and the scatter in their margin to dryout, the statistic η_{95} may be either positive or negative. A positive value indicates the setpoints determined using best-estimate simulation should be decreased by an appropriate amount to obtain a 95/95 result, while a negative value indicates that the best-estimate simulations are likely to under predict the true required setpoint due to the tendency of the minimum margin to trip to be underestimated (i.e., due to participants).

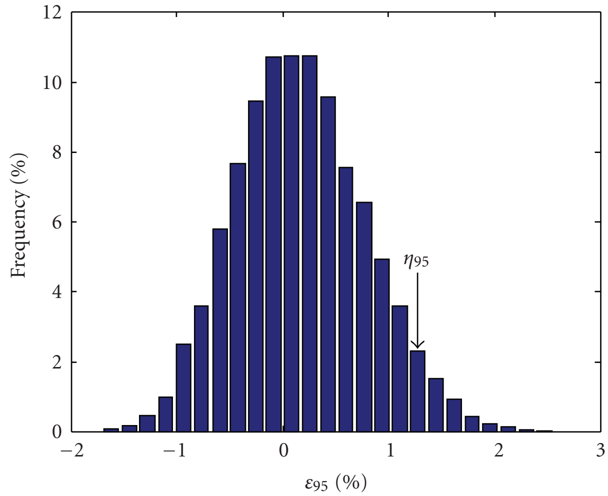

Figure 3: Distribution of 95th percentile trip setpoint errors over all core states.

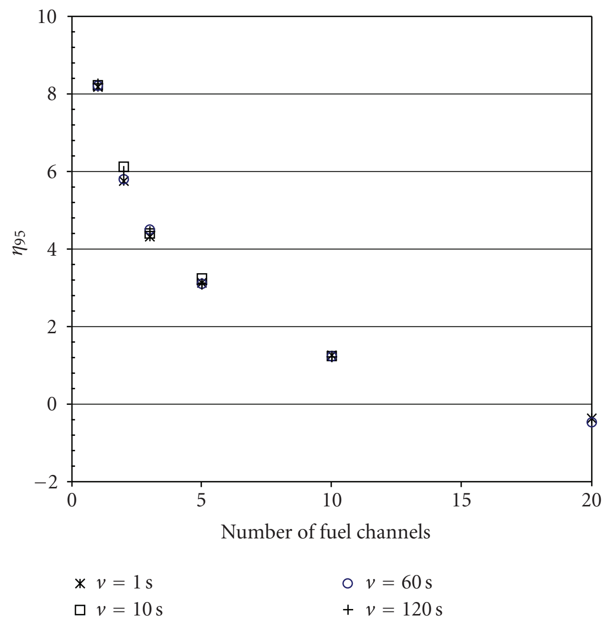

Figure 4: Allowance factor as a function of fuel channels and the transient accident speed.

4.3. *Sensitivity to power transients*

Figure 4 shows the trend in η_α as a function of the number of fuel channels considered in the demonstration. This is equivalent to considering situations where the core has less participants (i.e., core configurations that have outliers with margins to dryout substantive less than the surrounding fuel channels). This figure shows that for core states where outliers are a concern the compliance allowance factor increases. This is expected since the participation effect is reduced, and there is a smaller probability that other fuel channels may compensate for errors in the estimates of an outlier. (An alternative method for examining the effects of outliers would be to increase the distribution in the true channel powers and assess the impact on the uncertainty allowance.)

The effect of different exponential power transients is also shown in Figure 4 for exponential time constants of 1 second, 10 seconds, 60 seconds, and 120 seconds as a function of the number of fuel channels participating. The results show that the allowance factor becomes negative as the number of participating channels increases towards 20 (i.e., the best-estimate simulations themselves will provide at least a 95% probability and level of confidence). Furthermore, Figure 5 shows the behavior of the allowance factor for increasing numbers of participating detectors and for various power transient time constants. From Figures 4 and 5 it can be concluded that the allowance factor is not sensitive to the transient power rate (The changes in the allowance factor are

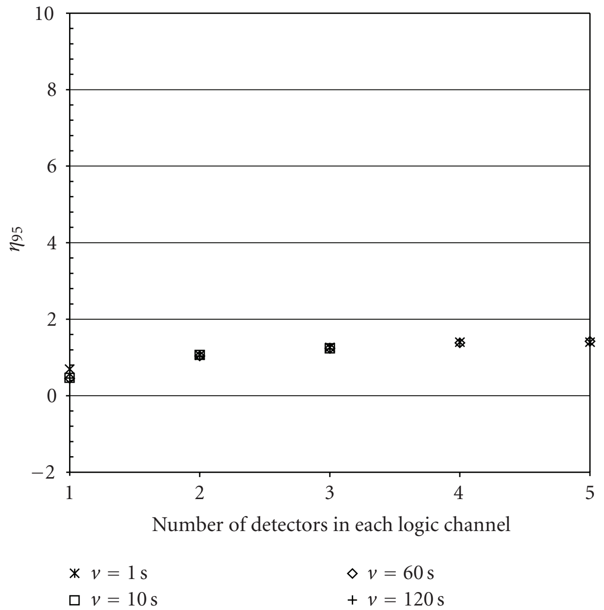

FIGURE 5: Allowance factor behavior as a function of the number of detectors in each logic channel and as a function of transient speed.

within the numerical accuracy of the Monte-Carlo simulations). It is an encouraging result of this methodology that the allowance factor is not significantly affected by the speed of the transient being considered, at least for the stylized LOR considered in this work.

5. CONCLUSIONS

A methodology for computing 95/95 trip setpoints for transient nuclear safety analysis has been presented which utilizes estimates over all fuel channels and detectors in a reactor core, and hence the errors in the maxima and minima predictions can be estimated. These estimates are used to ensure that there is a high probability and confidence that the acceptance criteria will be met for an accident. The methodology developed above represents a unique application of uncertainty analysis for estimation of setpoint errors required for safety analysis.

The statistical properties of the margin to dryout and margin to trip are separately investigated and in particular the behavior of the minimum estimated margin to trip and minimum margin to dryout are discussed. In general, it was observed that the number of fuel channels and detectors simulated impact the error observed in estimating the maxima or minima. These concepts were then applied to a hypothetical reactor transient involving a bulk power excursion event. Based on these simulations, the statistic used to correct the best estimates in trip setpoint was determined based upon the methodology outlined in this paper. For the hypothetical accident, the statistic decreases with increasing number of fuel channels and decreasing number of detectors. Furthermore, it has been demonstrated that the allowance factor increases only slightly with faster transients.

Finally, it is strongly recommended that for any best-estimate analysis, all fuel channels and detectors are appropriately modeled, or alternatively a group of channels where the minimum margin to dryout may occur and most probable tripping detectors must be considered in order to capture the true probabilities related to accident consequences. Furthermore, while this paper examined the margin to dryout behavior for a CANDU pressurized heavy water reactor, the results may be adopted for LWR analyses provided that the required margin to DNB is used.

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Research Article

GRS Method for Uncertainty and Sensitivity Evaluation of Code Results and Applications

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Recommended by Cesare Frepoli

During the recent years, an increasing interest in computational reactor safety analysis is to replace the conservative evaluation model calculations by best estimate calculations supplemented by uncertainty analysis of the code results. The evaluation of the margin to acceptance criteria, for example, the maximum fuel rod clad temperature, should be based on the upper limit of the calculated uncertainty range. Uncertainty analysis is needed if useful conclusions are to be obtained from "best estimate" thermal-hydraulic code calculations, otherwise single values of unknown accuracy would be presented for comparison with regulatory acceptance limits. Methods have been developed and presented to quantify the uncertainty of computer code results. The basic techniques proposed by GRS are presented together with applications to a large break loss of coolant accident on a reference reactor as well as on an experiment simulating containment behaviour.

1. INTRODUCTION

Best estimate computer codes are used to calculate postulated loss of coolant accidents and transient in a realistic way and not in a conservative way. There is an increasing interest in computational reactor safety analysis to replace the conservative evaluation model calculations by best estimate calculations supplemented by a quantitative uncertainty analysis. The USA Code of Federal Regulation (CFR) 10 CFR 50.46 [1], for example, allows either to use a best estimate code plus identification and quantification of uncertainties, or the conservative option using conservative computer code models listed in Appendix K of the CFR, Title 10, Part 50.

Code predictions are uncertain due to several sources of uncertainty, like code models as well as uncertainties of plant and fuel parameters. These uncertainties, for example, come from scatter of measured values, approximations of modelling, variation and imprecise knowledge of initial and boundary conditions. Computer code models are developed based on experiments which can simulate the complex behaviour of a reactor plant under accident conditions in a simplified way only. Most of the experiments are performed in small scale compared to plant size. Uncertainty due to imprecise knowledge of parameter values in calculations is quantified by ranges and probability distributions. These distributions should be taken into account for input parameters instead of one discrete value only.

Stochastic variability due to possible component failures of the reactor plant is not considered in an uncertainty analysis. The single failure criterion is still taken into account in a deterministic way. This is a superior principle of safety analysis and requirements of redundance. The probability of system failures is part of probabilistic safety analyses, not of demonstrating the effectiveness of emergency core cooling systems.

The aim of the uncertainty analysis is at first to identify and quantify all potentially important uncertain parameters. Their propagation through computer code calculations provides probability distributions and ranges for the code results. The evaluation of the margin to acceptance criteria, for example, the maximum fuel rod clad temperature, should be based on the upper limit of this distribution for the calculated temperatures, see Figure 1. Uncertainty analysis is needed if useful conclusions with regard to prediction capability, such as maximum cladding temperature, are to be obtained from "best estimate" thermal-hydraulic code calculations, otherwise single values of unknown accuracy would be presented for comparison with limits for acceptance.

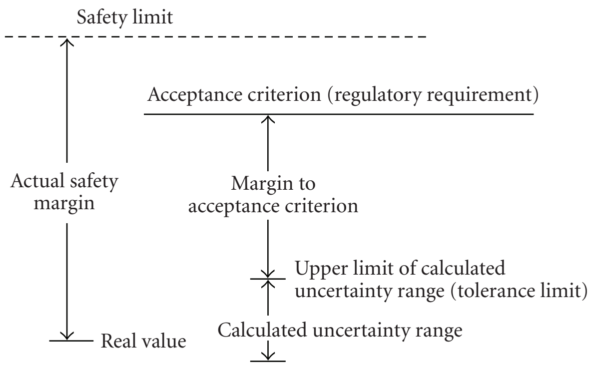

Figure 1: Margin illustration.

Section 2 describes the GRS method, Section 3 presents examples of application of the GRS method, and Section 4 provides conclusions.

2. DESCRIPTION OF THE GRS METHOD

Among others, GRS method [2] has been developed for the determination of uncertainties. The state of knowledge about all uncertain parameters is described by ranges and probability distributions, Figure 2. In order to get information about the uncertainty of computer code results, a number of code runs have to be performed. For each of these calculation runs, all identified uncertain parameters are varied simultaneously. Uncertain parameters are uncertain input values, models, initial and boundary conditions, numerical values like convergence criteria and maximum time step size, and so forth. Model uncertainties are expressed by adding on or multiplying correlations by corrective terms, or by a set of alternative model formulations. Uncertainties in noding, to describe the important phenomena, are to be taken into account in the code validation process. However, alternative noding schemes can be included in the uncertainty analysis. Code validation results are a fundamental basis to quantify parameter uncertainties.

The selection of parameter values according to their specified probability distributions, their combination, and the evaluation of the calculation results requires a method. Following a proposal by GRS, the central part of the method is a set of statistical techniques. The advantage of using these techniques is that the number of code calculations needed is independent of the number of uncertain parameters. In each code calculation, all uncertain parameters are varied simultaneously. In order to quantify the effect of these variations on the result, statistical tools are used. Because the number of calculations is independent of the number of uncertain parameters, no a priori ranking of input parameters is necessary to reduce their number in order to cut computation cost. The ranking is a result of the analysis as described later.

The number of code calculations depends on the requested probability content and confidence level of the statistical tolerance limits used in the uncertainty statements of

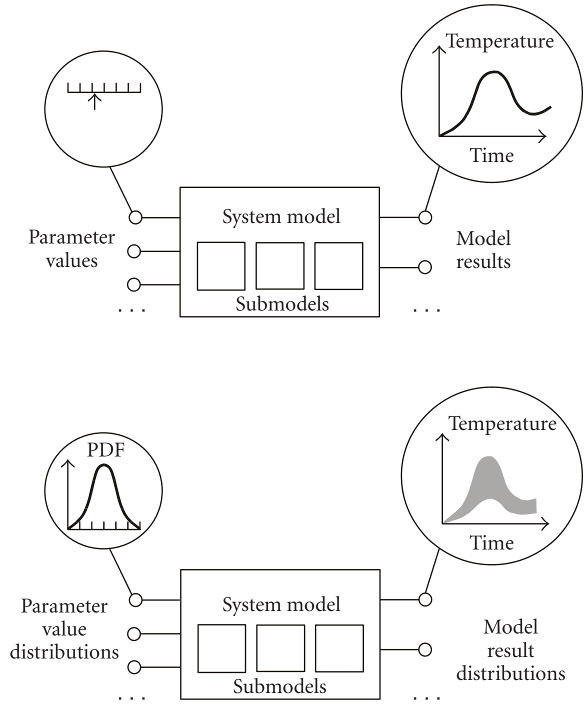

Figure 2: Consideration of input parameter value ranges instead of discrete values in the GRS method.

the results. The required minimum number n of these calculation runs is given by Wilks' formula [3, 4], for example, for one-sided tolerance limits: $1 - a^n \geq b$, where $b \times 100$ is the confidence level (%) that the maximum code result will not be exceeded with the probability $a \times 100$ (%) (percentile) of the corresponding output distribution, which is to be compared to the acceptance criterion. The confidence level is specified to account for the possible influence of the sampling error due to the fact that the statements are obtained from a random sample of limited size. For two-sided statistical tolerance intervals, the formula is: $1 - a^n - n(1-a)a^{n-1} \geq b$. The minimum number of calculations can be found in Table 1.

The probabilistic treatment of parameter uncertainties allows quantifying their state of knowledge. This means, in addition to the uncertainty range, the knowledge is expressed by probability density functions or probability distributions. This interpretation of probability is used for a parameter with a fixed but unknown or inaccurately known value. The classical interpretation of probability as the limit of a relative frequency, expressing the uncertainty due to stochastic variability, is not applicable here.

The probability distribution can express that some values in the uncertainty range are more likely to be the appropriate parameter value than others. In the case that no preferences can be justified, uniform distribution will be specified, that is, each value between minimum and maximum is equally likely to be the appropriate parameter value. As the consequence of this specification of probability distributions of input parameters, the computer code results also show a

Table 1: Minimum number of calculations n for one-sided and two-sided statistical tolerance limits.

	One-sided statistical tolerance limits			Two-sided statistical tolerance limits		
b/a	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

probability distribution, from which uncertainty limits or intervals are derived.

A total number of n code runs are performed varying simultaneously the values of all uncertain input parameters, according to their distribution. The n values of the considered output parameters are ordered: $Y(1) < Y(2) \cdots < Y(n-1) < Y(n)$. Therefore, the name-order statistics is used for Wilks' formula. On the basis of this ranking, the 95th percentile value with a confidence level of 95% is obtained by selecting $Y(n)$ with $n = 59$ for the one-sided tolerance limit, for example. A 5th percentile value with a confidence level of 95% is obtained by selecting $Y(1)$ with $n = 59$. A (95%/95%) two-sided tolerance limit is obtained by selecting $Y(1)$ and $Y(n)$ with $n = 93$.

Another important feature of the method is that one can evaluate sensitivity measures of the importance of parameter uncertainties for the uncertainties of the results. These measures give a ranking of input parameters. This information provides guidance as to where to improve the state of knowledge in order to reduce the output uncertainties most effectively, or where to improve the modelling of the computer code. Sensitivity measures like standardised rank regression coefficients, rank correlation coefficients, and correlation ratios permit a ranking of uncertainties in model formulations, input data, and so forth, with respect to their relative contribution to code output uncertainty. The difference to other known uncertainty methods, for example, [5], is that the ranking is a result of the analysis and not of prior estimates and judgements. This prior setup of a phenomena identification and ranking table (PIRT) by extensive expert staff-hours in [5] is known to be very costly. Uncertainty statements and sensitivity measures are available simultaneously for all single-valued (e.g., peak clad temperature) as well as continuous valued (time dependent) output quantities of interest. The method relies only on actual code calculations without using approximations like fitted response surfaces. Similar methods based on the GRS method, and an alternative uncertainty method is presented in [6].

The different steps of the uncertainty analysis according to the GRS method are supported by the software system for uncertainty and sensitivity analyses (SUSA) developed by GRS [7]. They provide a choice of statistical tools to be applied during the uncertainty and sensitivity analysis.

3. APPLICATIONS

The GRS method for uncertainty and sensitivity evaluation of code results can be used for different codes to investigate the combined influence of all potentially important uncertainties on the calculation results. Several applications have been performed in GRS to investigate loss of coolant from the primary and secondary coolant systems of pressurised water reactors, as well as related experiments. For these analyses, we used the thermal-hydraulic computer code ATHLET. Another uncertainty and sensitivity analysis was performed calculating an experiment simulating containment behaviour using the computer code COCOSYS.

3.1. *Thermal-hydraulic applications using the ATHLET computer code*

Several uncertainty and sensitivity analyses were performed by GRS using the thermal-hydraulic computer code ATHLET simulating breaks of the primary and secondary side cooling systems of pressurised water reactors. These are

(i) separate effects experiment OMEGA heater rod bundle Test 9,
(ii) integral experiment LSTF-CL-18, 5% cold leg break, accumulator injection into cold legs,
(iii) PWR 5% cold leg break, accumulator injection into hot legs (Siemens/ KWU reactor),
(iv) integral experiment LOFT L2-5, $2 \times 100\%$ cold leg break, accumulator injection into cold legs,
(v) PWR $2\times100\%$ cold leg break, combined ECC injection into cold and hot legs,
(vi) PWR 10% steam line break,
(vii) PSB-VVER 11% upper plenum break experiment, UP-11-08 (OECD PSB-VVER Test1).

One out of these applications is described in the following section.

3.2. *Application to a German PWR reference reactor,* $2 \times 100\%$ *cold leg break*

A double ended cold leg offset shear break design basis accident of a German PWR of 1300 MW electric power is investigated. The fuel rod peak linear heat generation rate is 530 W/cm. Loss of off-site power at turbine trip 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 for ECC injection from accumulator, high- and low- pressure system, and one hot leg accumulator is unavailable due to preventive maintenance. These assumptions are considered to be the worst unavailability, agreed between applicants and assessors.

The uncertainty analysis considered 56 uncertain input parameters. These consist of 37 model parameters, 4 parameters to select different model correlations for heat transfer and friction, 2 for bypass flow cross sections in the reactor vessel, 1 for temperature of accumulator water, 1 for core power, 1 for decay heat, 1 for radial power distribution in the core, 1 for hot channel factor, 5 for gap width (5 burn-up classes), 1 for fuel thermal conductivity, and 2 for convergence criteria. The model parameters comprise critical flow, heat transfer, evaporation, condensation, wall and interfacial shear, form loss, main coolant pump head, and torque.

A total number of 100 calculations were performed using the code ATHLET Mod 1.2, cycle D [8].

3.3. Maximum clad temperature

Figure 3 shows at any point of time, at least 95% of the combined influence of all considered uncertainties on the calculated clad temperatures is below the presented uncertainty limit (one-sided tolerance limit), at a confidence level of at least 95%. For each instant of time, the desired tolerance limits were selected from the 100 calculated code results. A "conservative" calculation result is shown for comparison, applying the best estimate code ATHLET with default values of the models and conservative values for the initial and boundary conditions reactor power, decay heat, gap width of fuel rods between fuel and clad, fuel pellet thermal conductivity, and temperature of accumulator water. All these conservative values were also included in the distributions of the input parameters for the uncertainty analysis. The maximum clad temperature of the conservative calculation does not bound the 95%/95% one-sided tolerance limits of the uncertainty analysis over the whole transient time, for example, after 75 seconds. The regulatory acceptance criterion for peak clad temperature is 1200°C.

The "conservative" calculation is representative for the use of best estimate computer codes plus conservative initial and boundary conditions. Such an evaluation is possible in the licensing procedure of several countries, but not in the USA. The uncertainty of code models is not taken into account. The selection of conservative initial and boundary conditions will bound these model uncertainties. That is obviously not the case for the whole transient in the present example. An uncertainty analysis quantifies uncertain initial and boundary conditions as well as model uncertainties. The peak clad temperatures, however, are bounded due to cumulating conservative values of the highly sensitive parameters gap width and pellet thermal conductivity. It is obvious that the results are dependent on the extent of conservatism implemented in the conservative calculations. Therefore, the US Code of Federal Regulation [1] 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 best-estimate computer code is used for the analysis.

According to the US Code of Federal Regulations, Title 10, Section 50.46, the conservative method requires conservative models to be applied in conformity with the required and acceptable features listed in Appendix K, "ECCS Evaluation Models" of the Federal Regulations [1]. This is the main reason why, in the USA, an additional margin to licensing criteria is available by changing from conservative evaluation to best estimate calculations plus uncertainty analysis.

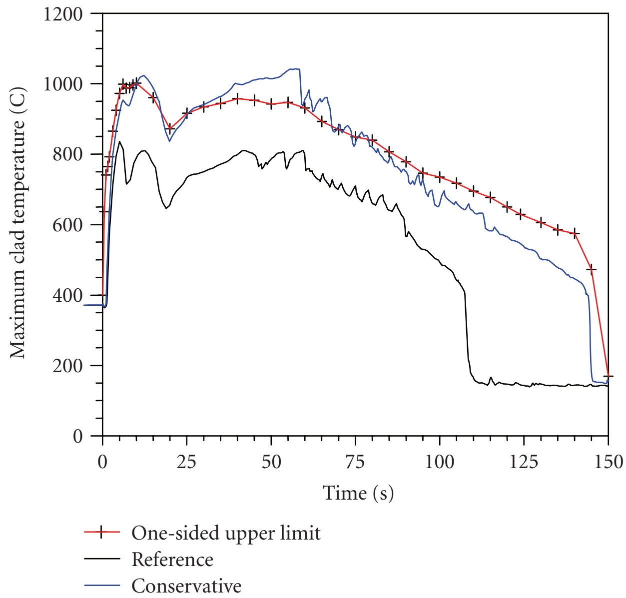

Figure 3: Calculated one-sided 95%/95% uncertainty limit and best estimate reference calculation compared with a "conservative" calculation of rod clad temperature for a reference reactor during a postulated double ended offset shear cold leg break.

The confidence level 95% denominates that the 95th percentile is overestimated conservatively by 95% probability providing a (95%, 95%) statement. This conservatism is the reason why some experts claim that a coverage of a (95%, 95%) statement by a conservative calculation is not needed. GRS requires coverage unless other suitable methods for comparison and quantification of "conservatisms" are presented. This could be achieved by an additional statistical test proving that the conservative calculation bounds the 95th percentile.

3.3.1. Sensitivity measures

Sensitivity measures indicate the influence of the uncertainty in input parameters on calculation results. For example, the Spearman rank correlation coefficient is used as sensitivity measure. The length of the bars indicates the sensitivity of the respective input parameter uncertainty on the first peak clad temperature which occurs during the blowdown phase; see Figure 4. The sensitivity measure gives the variation of the result in terms of standard deviations when the input uncertainty varies by one standard deviation (if the input uncertainties are independent). Positive sign means that input parameter value and result tend to move in the same direction, that is, an increase of uncertain input parameter value

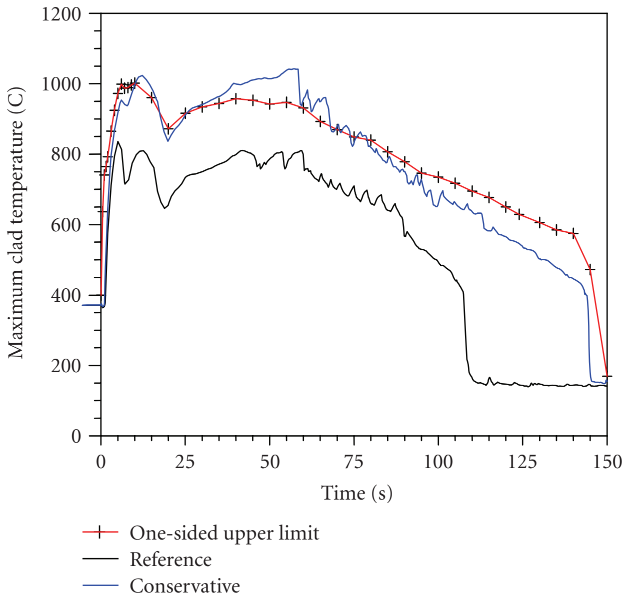

FIGURE 4: Sensitivity measures of the blowdown PCT with respect to the selected 56 uncertain input parameters (rank correlation coefficient) for the reference reactor large break.

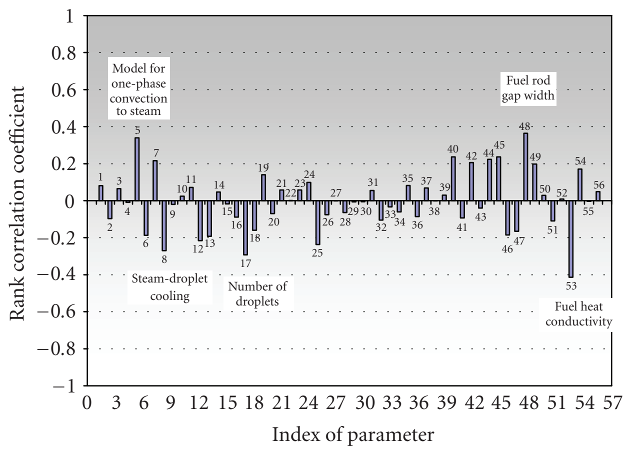

FIGURE 5: Sensitivity measures of the reflood PCT with respect to the selected 56 uncertain input parameters (rank correlation coefficient) for the reference reactor large break.

tends to increase the clad temperature and vice versa. For negative sign, the input parameter value and the result tend to move in opposite direction, that is, an increase of the parameter value tends to decrease the clad temperature and vice versa.

The most important parameter uncertainties, out of 56 identified potentially important parameters, with respect to the blowdown peak clad temperature uncertainty are

(i) fuel rod gap width for low burn up (positive sign),
(ii) fuel heat conductivity (negative sign),
(iii) minimum film boiling temperature (negative sign),
(iv) model for critical heat flux (negative sign: Biasi correlation causes lower clad temperatures due to a later change from nucleate to transition boiling compared to the Hench-Levy correlation),
(v) reactor initial power (positive sign),
(vi) 2-phase multiplier in horizontal pipe (negative sign: higher resistance of water transport to break location ⇒higher water content in core due to lower break flow ⇒lower clad temperature).

The most important parameters for the peak clad temperature uncertainty during reflood are, according to Figure 5,

(i) fuel heat conductivity (negative sign),
(ii) fuel rod gap width for low burn up (positive sign),
(iii) model for 1-phase convection to steam (positive sign, i.e., Mc Eligot correlation tends to cause higher clad temperatures than Dittus-Boelter II),
(iv) number of droplets (negative sign: number of droplets higher ⇒higher condensation ⇒lower PCT),
(v) steam-droplet cooling (negative sign: higher cooling tends to result in lower PCT).

3.4. Application to the experiment HDR T31.5 simulating containment behaviour

The experiment T31.5 on the HDR containment facility simulates a large break of a main coolant pipe, investigating steam and gas release into the containment according to the low pressure scenario of the German risk study. A short term phase was performed with emphasis on pressure buildup in the containment and the temperature evolution.The hydrogen distribution was measured during a long term phase over 20 hours, when steam and a helium-hydrogen mixture were injected.

A total number of 200 calculations were performed using the code COCOSYS V0.2 [9]. At least 95% of the combined influence of all considered uncertainties on the calculated pressure at a confidence level of at least 95% at any point of time is shown in Figure 6. A total of 79 uncertain parameters were included, consisting of model parameters, of the experimental facility, initial and boundary conditions.

Sensitivity measures about the influence of the uncertainty in input parameters on the pressure in the upper part of the HDR containment versus time are presented in Figure 7. We see decreasing and increasing high importance versus time on the maximum pressure. Decreasing influence with time is due to decreasing energy transport with decreasing convection for

(i) free convection, parameter 72, negative sign,
(ii) forced convection, parameter 73, negative sign,
(iii) condensation at wall, parameter 74, negative sign.

Increasing with time are the following parameters because of decreasing convection:

(i) thickness of liner, parameter 79, negative sign,
(ii) surface of liner, parameter 77, negative sign,
(iii) heat capacity of concrete structures, parameter 69, negative sign.

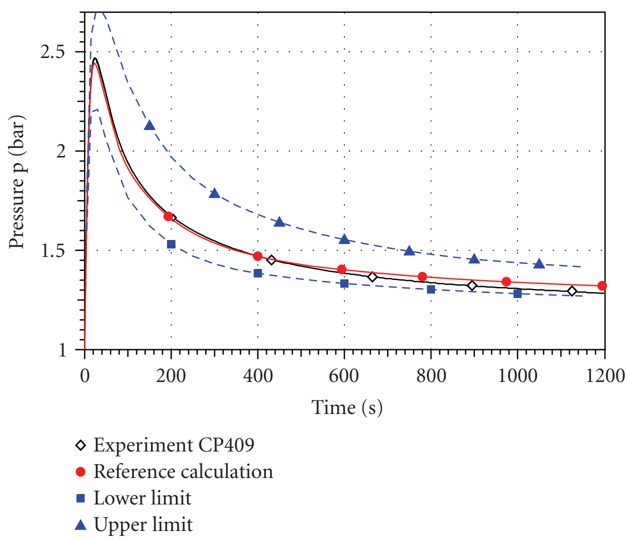

Figure 6: 95%/95% uncertainty interval, reference calculation and experimental values for pressure in the upper part of the containment versus time.

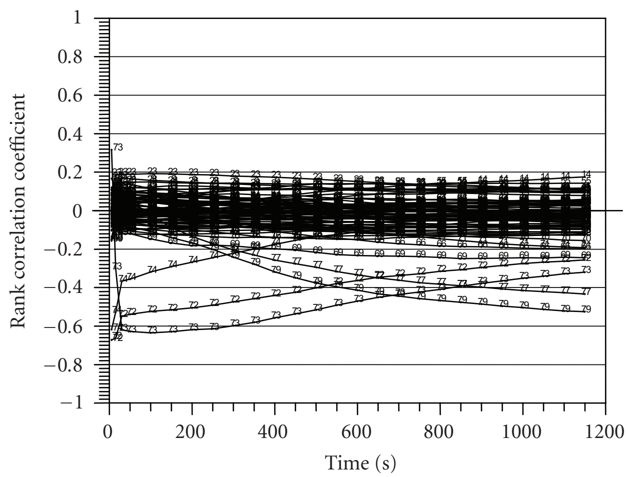

Figure 7: Sensitivity measures for pressure in the upper part of the containment versus time.

4. CONCLUSIONS

Two applications of the uncertainty method proposed by GRS are presented. A significant advantage of this methodology is that no a priori reduction in the number of uncertain input parameters by expert judgement or screening calculations is necessary to limit the calculation effort. All potentially important parameters may be included in the uncertainty analysis. The method accounts for the combined influence of all identified input uncertainties on the results. This would be difficult or even impossible to achieve by a priori expert judgement of loss of coolant accidents or transients.

The number of calculations needed is independent of the number of uncertain parameters accounted for in the analysis. It does, however, depend on the requested tolerance limits, that is, the requested probability coverage (percentile) of the combined effect of the quantified uncertainties, and on the requested confidence level of the code results. The tolerance limits can be used for quantitative statements about margins to acceptance criteria.

Another important feature of the method is that it provides sensitivity measures of the influence of the identified input parameter uncertainties on the results. The measures permit an uncertainty importance ranking. This information provides guidance as to where to improve the state of knowledge in order to reduce the output uncertainties most effectively, or where to improve the modelling of the computer code. Different to other known uncertainty methods, the ranking is a result of the analysis and its inputs and not of an a priori expert judgement. Uncertainty statements and sensitivity measures are available simultaneously for all single-valued (e.g., peak cladding temperature) as well as continuous valued (time dependent) output quantities of interest. The method relies only on actual code calculations without the use of approximations like fitted response surfaces. The method proposed by GRS has been used in different applications by various international institutions including licensing.

A challenge in performing uncertainty analyses is the specification of ranges and probability distributions of input parameters. Investigations are underway to transform data measured in experiments and post test calculations into thermal-hydraulic model parameters with uncertainties. Care must be taken to select suitable experimental and analytical information to specify uncertainty distributions. This is a general experience gained in applying different uncertainty methods.

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Review Article

International Course to Support Nuclear Licensing by User Training in the Areas of Scaling, Uncertainty, and 3D Thermal-Hydraulics/Neutron-Kinetics Coupled Codes: 3D S.UN.COP Seminars

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Thermal-hydraulic system computer codes are extensively used worldwide for analysis of nuclear facilities by utilities, regulatory bodies, nuclear power plant designers, vendors, and research organizations. The computer code user represents a source of uncertainty that can influence the results of system code calculations. This influence is commonly known as the "user effect" and stems from the limitations embedded in the codes as well as from the limited capability of the analysts to use the codes. Code user training and qualification represent an effective means for reducing the variation of results caused by the application of the codes by different users. This paper describes a systematic approach to training code users who, upon completion of the training, should be able to perform calculations making the best possible use of the capabilities of best estimate codes. In other words, the program aims at contributing towards solving the problem of user effect. In addition, this paper presents the organization and the main features of the 3D S.UN.COP (scaling, uncertainty, and 3D coupled code calculations) seminars during which particular emphasis is given to the areas of the scaling, uncertainty, and 3D coupled code analysis.

1. INTRODUCTION

A wide range of activities has recently been completed in thearea of system thermal-hydraulics as a follow-up to considerable research efforts. Problems have been addressed, solutions to which have been at least partly agreed upon on international ground. These include the need for best-estimate system codes [1, 2], the general code qualification process [3, 4], the proposal for nodalization qualification, and attempts aiming at qualitative and quantitative accuracy evaluations [5]. Complex uncertainty methods have been proposed, following a pioneering study at USNRC [6]. This study attempted, among other things, to account for user effects (see Section 2 for definition) on code results. An international study aiming at thecomparison of assumptions and results of code uncertainty methodologies has been completed [7].

More recently (during the period 1997–1999), the IAEA (International Atomic Energy Agency) developed a document consistent with its revised Nuclear Safety Standards Series [8] that provides guidance on accident analysis of nuclear power plants (NPPs). The report includes a number of practical suggestions on the manner in which to perform accident analysis of NPPs. These cover the selection of initiating events, acceptance criteria, computer codes, modeling assumptions, the preparation of input, qualification of users, presentation of results, and quality assurance. The suggestions are both conceptual as well as formal and are based on present practice worldwide for performing accident analysis. The report covers all major steps in performing analyses and is intended primarily for code users.

Within the framework of the "Nuclear Safety Standard Series" the important role of the user's effects on the analysis has been addressed. The need for user qualification and training has been clearly recognized and the systematic training of analysts was emphasized as being crucial for the quality of the analysis results. Three areas of training, in particular, have been specified in the following:

(i) practical training on the design and operation of the plant;
(ii) software specific training;
(iii) application specific training.

Training on the phenomena and methodologies is typically provided at the university level, but cannot always be considered sufficient. Furthermore, training on the specific application of system codes is not usually provided at this level, whereas practical training on the design and operation of the plant is essential for the development of the plant models. Software specific training is important for the effective use of the individual code. Application specific training requires the involvement of a strong support group that shares its experience with the trainees and provides careful supervision and review. Training at all three levels ending with examination is encouraged for a better effectiveness of the training. Such a procedure is considered a step in the direction of establishing a standard approach that could be applicable to an international basis.

Based on the above considerations and facts, the paper outlines the role of the code user, addresses the problem of the user's effect in Section 2, provides a proposal for a permanent training course for system codes in Section 3, and gives a tangible example of user-training-course (i.e., 3D S.UN.COP), mostly focused on the development and application of best-estimate codes emphasizing scaling, best-estimate, uncertainty, and 3D coupled code calculations analyses, in Section 4.

2. THERMAL-HYDRAULIC CODES AND CODE USERS

2.1. Role and relevance of code user

The best estimate thermal-hydraulic codes used in the area of nuclear reactor safety have reached a marked level of sophistication. Their capabilities to predict accidents and transients at existing plants have substantially improved over the past years as a result of large research efforts and can be considered satisfactory for practical needs provided that they are used by competent analysts.

Best estimate system codes (RELAP, TRAC, CATHARE, or ATHLET) are currently used by designer/vendors of NPPs, by utilities, licensing authorities, research organizations including universities, nuclear fuel companies, and by technical support organizations. The objectives of using the codes may be quite different, ranging from design or safety assessment to simply understanding the transient behavior of a simple system. However, the ap-plication of a selected code must be proven to be adequate to the performed analysis. Many aspects from the design data necessary to create input to the selection of the noding solutions and the code itself are the user's responsibility [9–11].

The role of the code user is extremely relevant: experience with large number of International Standard Problems (ISPs) has shown the dominant influence of the code user on the final results and the goal of reduction of user effects has not been achieved. It has been observed previously that

(i) the user gives a contribution to the overall uncertainty that unavoidably characterizes system code calculation results;
(ii) in the majority of cases, it is impossible to distinguish among uncertainty sources like "user effect," "nodalization inadequacy," "physical model deficiencies," "uncertainty in boundary or initial conditions," and "computer/compiler effect;"
(iii) "reducing the user effect" or "finding the optimum nodalization" should not be regarded as a process that removes the need to assess the uncertainty.

Performing an adequate code analysis or assessment involves two main aspects.

(1) *Code adequacy*. The adequacy demonstration process must be undertaken by a code user when a code is used outside its assessment range, when changes are made to the code, and when a code is used for new applications where different phenomena are expected. The impact of these changes must be analyzed and the analyses must be thoroughly reviewed to ensure that the code models are still adequate to represent the phenomena that are being observed.
(2) *Quality of results*. Historically the results of code predictions, specifically when compared with experimental data gathered from applicable scaled test facilities, have revealed inadequacies raising concerns about code reliability and their practical usefulness. Discrepancies between measured and calculated values were typically attributed to model deficiencies, approximation in the numeric solutions, computer, and compiler effects,nodalization inadequacies, imperfect knowledge of boundary and initial conditions, unrevealed mistakes in the input deck, and to "user effect." In several ISPs sponsored by OECD (Organisation for Economic Cooperation and Development), several users modeled the same experiment using the same code, and the code-calculated results varied widely, regardless of the code used. Some of the discrepancies can be attributed to the code user approach as well as to a general lack of understanding of both the facility and the test.

The two items are the main aspects, both related to the code user. The first aspect is included in the qualification framework of the code and nodalization. The second aspect is directly related to the user choices generally referred to as User Effect.

2.2. User effect

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

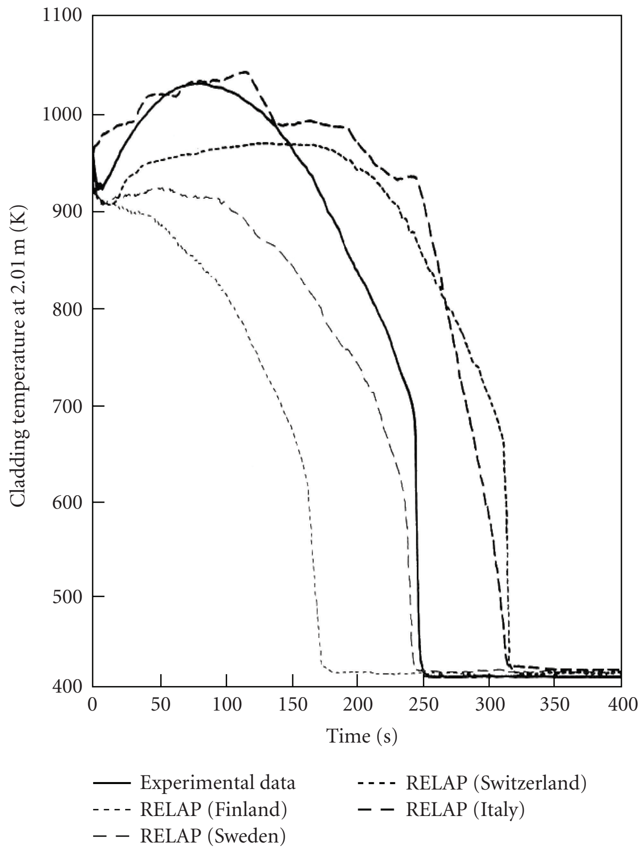

FIGURE 1: User effect: different results for the cladding temperature in the ISP25 test from different users adopting the same code and BIC.

component). In addition, even two competent users will not approach the analysis of a problem in the same way and consequently, will likely take different paths to obtain a problem solution. The cumulative effect of user community members to produce a range of answers using the same code for a well-defined problem with rigorously specified boundary and initial conditions is the user effect (see Figure 1).

The following are some of the reasons for the user effects.

(i) Code use guidelines are not fully detailed or comprehensive.

(ii) Based on the current state of the art, the actual 3D plant geometries are usually modeled using several 1D zones; these complex 3D geometries are suitable for different modeling alternatives; as a consequence an assigned reactor vessel part is modeled differently by different users of the same code. Beside the major 1dimensional code modules, a number of empirical models for system components, such as pumps, valves, and separators, are specified by the users, sometimes based on extrapolation from scaled devices, thereby introducing additional inaccuracies.

(iii) Experienced users may overcome known code limitations by adding engineering knowledge to the input deck.

(iv) Problems inherent to a given code or a particular facility have been dealt with over the years by the consideration and modeling of local pressure drop coefficients, critical flow rate multipliers, or other bias to obtain improved solutions. This has been traditionally done to compensate for code limitation (e.g., application of steady-state qualified models to transient conditions, and lack of validity of the fully developed flow concept in typical nuclear reactor conditions). Furthermore, specific effects such as small bypass flows or distribution of heat losses might exacerbate the user effect.

(v) The increasing number of users performing analysis with insufficient training. As such, their lack of understanding of the code capabilities and limitations leads to incorrect interpretation of results. The failure to obtain a stable steady state by the user prior to the initiation of the transient is included in this item.

(vi) A nonnegligible effect on code results comes from the compiler and the computer used to run an assigned code selected by the user; this remains true for very recent code versions.

(vii) Error bands and the values of initial and boundary conditions which are needed as code inputs are not well defined; this ambiguity is used to justify inappropriate model modifications or interpretation of results.

(viii) Analysts lack complete information about facilities before developing input and hence filling the gaps with unqualified data.

(ix) Although the number of user options is thought to be reduced in the advanced codes, for some codes there are several models and correlations for the user to choose. The user is also required to specify parameters such as pressure loss coefficients, manometric characteristics, efficiencies, and correlation factors which may not be well defined.

(x) Most codes have algorithms to adjust the time step control (e.g., Courant limit) to maximum efficiency and minimize run time. However, users are allowed to change the time step to overcome code difficulties and impose smaller time steps for a given period of the transient. If the particular code uses an explicit numerical scheme, the result will vary significantly with the time step size.

(xi) Quality assurance guidelines should be followed to check the correctness of the values introduced in the input despite the automatic consistency checks provided by the code.

Typical examples of user and other related effects on code calculations of selected experiments are presented in several CSNI reports (e.g., ISP-25, ACHILLES reflooding test; LOBI natural circulation test; ISP-22 on SPES Loss-Of-Feed-Water test; ISP-26 on LSTF 5% cold-leg-break loss-of-coolant-accident (LOCA); ISP-27 on BETHSY 2” cold-leg LOCA) and based on these outcomes different organizations have defined in what follows some general principles in order to reduce the user effects.

(i) The misapplication of the system code should be eliminated (or reduced at least) by means of sufficiently detailed code description and by relevant code user guidelines.

(ii) 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.

(iii) The user community should preferably use the same computing platform (i.e., the machine round-off errors and treatment of arithmetic operations are assumed the same).

(iv) The system code should preferably be used by a relatively large user community (a large sample size).

(v) The problem to be analyzed should be rigorously specified (i.e., all geometrical dimensions, initial conditions, and boundary conditions should be clearly specified).

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

(i) the flexibility of the system code. An example is the flexibility associated with modeling a system component such as the steam generator: for instance, 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 and BRANCH; consequently, there are more degrees of freedom available to the user, each requiring a decision, when a RELAP5 steam generator model is being constructed than when a TRAC-generated model of the same component is being defined;

(ii) the practices used to define the nodalization and to ensure that a convergent solution is achieved. In this context, the code validation process, the nodalization qualification, and the qualitative or quantitative accuracy evaluation are necessary steps to reduce the possibility of producing poor code predictions [12, 13].

3. PERMANENT USER TRAINING COURSE FOR SYSTEM CODE: THE PROPOSAL

As a follow-up to the specialists meeting held at the IAEA in September 1998, the Universities of Pisa and Zagreb and the Jožef Stefan Institute, Ljubljana, jointly presented a Proposal to IAEA for the Permanent Training Course for System Code Users [14]. It was recognized that such a course would represent both a source of continuing education for current code users and a means for current code users to enter the formal training structure of a proposed "permanent" stepwise approach to user training.

As a follow-up to the massive work conducted in different organizations, the need was felt to fix criteria for training the code user. As a first step, the kind of code user and the level of responsibility of a calculation result should be discussed.

3.1. Levels of user qualification

Two main levels for code user qualification are distinguished in the following:

(i) code user, level "A" (LA);

(ii) responsible for the calculation results, level "B" (LB).

Two levels should be considered among LB code users to distinguish seniority (i.e., Level B, Senior (LBS)). Requisites are detailed hereafter for the LA grade only; these must be intended as a necessary step (in the future) to achieve the LB and the LBS grades. The main difference between LA and LB lies in the documented experience with the use of a system code; for the LB and the LBS grades, this can be fixed in 5 and 10 years, respectively, after achieving the LA grade. In such a context, any calculation having an impact in the sense previously defined must be approved by a LB (or LBS) code user and performed by a different LA or LB (or LBS) code user.

3.2. Requisites for code user qualification

3.2.1. LA code user grade

The identification of the requisites for a qualified code user derives from the areas and the steps concerned with a qualified system code calculation: a system code is one of the codes previously defined and a qualified calculation in principle includes the uncertainty analysis. The starting condition for LA code user is a scientist with generic knowledge of nuclear power plants and reactor thermal hydraulics (e.g., in possession of the master degree in US, of the "Laurea" in Italy, etc.).

The requisites competencies for the LA grade code user are in the following areas.

(A) Generic code development and assessment processes:

Subarea (*A1*): conservation (or balance) equations in thermal hydraulics including definitions like HEM/EVET, UVUT(UP), Drift Flux, 1D, 3D, 1-field, Multifield, [2], conduction and radiation heat transfer, Neutron Transport Theory and Neutron Kinetics approximation, constitutive (closure) equations including convection heat transfer, special components (e.g., pump, separator), material properties, simulation of nuclear plant and BoP related control systems, numerical methods, general structure of a system code;

Subarea (*A2*): developmental assessment, independent assessment including Separate Effect Tests (SETF) Code Validation Matrix [3], and Integral Test (ITF) Code Validation Matrix [4]. Examples of specific Code validation Matrices.

(B) Specific code structure:

Subarea (*B1*): structure of the system code which is selected by the LA code user: thermal hydraulics, neutronics, control system, special components, material properties, numerical solution;

Subarea (*B2*): structure of the input; examples of user choices.

(C) Code use-Fundamental Problems (FP):

Subarea (*C1*): definition of the Fundamental Problem (FP): simple problems for which analytical solution may be available or less. Examples of code results from applications to FP; different areas of the code must be concerned (e.g., neutronics, thermal hydraulics, and numerics);

Subarea (*C2*): the LA code user must deeply analyze at least three specified FPs, searching for and characterizing the effects of nodalization details, time step selection and other code-specific features (to develop a nodalization starting from a supplied data base or problem specifications; to run a reference test case; to compare the results of the reference test case with data (experimental data, results of other codes, analytical solution), if available; to run sensitivity calculations; and to produce a comprehensive calculation report (having an assigned format).

(D) Basic Experiments and Test Facilities (BETF):

Subarea (*D1*): definition of BasicExperiments and test facilities (BETF): research aiming at the characterization of an individual phenomenon or of an individual quantity appearing in the code implemented equations, not necessarily connected with the NPP. Examples of code results from applications to BETF;

Subarea (*D2*): the LA code user must deeply analyze at least two selected BETF, searching for and characterizing the effects of nodalization details, time step selection, error in boundary and initial conditions, and other code-specific features.

(E) Code use-Separate Effect Test Facilities (SETF):

Subarea (*E1*): Definition of Separate Effect Test Facility (SETF): test facility where a component (or an ensemble of components) or a phenomenon (or an ensemble of phenomena) of the reference NPP is simulated. Details about scaling laws and design criteria. Examples of code results from applications to SETF;

Subarea (*E2*): The LA code user must deeply analyze at least one specified SETF experiment, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions, and other code-specific features.

(F) Code use-Integral Test Facilities (ITF):

Subarea (*F1*): definition of Integral Test Facility (ITF): test facility where the transient behavior of the entire NPP is addressed. Details about scaling laws and design criteria. Details about existing (or dismantled) ITF and related experimental programs. ISPs activity. Examples of code results from applications to ITF;

Subarea (*F2*): the LA code user must deeply analyze at least two specified ITF experiments, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions and other code-specific features.

(G) Code use-Nuclear Power Plant transient Data:

Subarea (*G1*): description of the concerned NPP and of the relevant (to the concerned NPP and calculation) BoP and ECC systems. Examples of code results from applications to NPP;

Subarea (*G2*): the LA code user must deeply analyze at least two specified NPP transients, searching for and characterizing the effects of nodalization details, time step selection, errors in boundary and initial conditions and other code-specific features.

(H) Uncertainty methods including concepts like nodalization, accuracy quantification, and user effects:

Subarea (*H1*): Description of the available uncertainty methodologies. The LA code user must be aware of the state of the art in this field.

3.2.2. LB code user grade

A qualified user at the LB grade must be in possession of the same expertise as the LA grade and

(i) he must have a documented experience in the use of system codes of at least 5 additional years;
(ii) he must know the fundamentals of Reactor Safety and Operation- and Design having generic expertise in the area of application of the concerned calculation;
(iii) he must be aware of the use and of the consequences of the calculation results; this may imply the knowledge of the licensing process.

3.2.3. LBS code user grade

A qualified user at the LBS grade must be in possession of the same expertise as the LB grade and

(i) he must have an additional documented experience in the use of system codes of at least 5 additional years. Moreover, the LBS code user is responsible for documenting user guidelines, methodology descriptions, and for providing technical leadership in R&D activities.

3.3. Course conduct and modalities for the achievements of code user grades

The training of the code user requires the conduct of lectures, practical on-site exercises, homework, and examination, while for the senior code user, only a review of documented experience and on-site examination is foreseen. The code user training, including practical exercises which represent an essential part of the course, lasts two years and covers the areas from (A) to (H).

The modalities defined in Table 1 are necessary to achieve the LA, LB (5 years after the LA grade), and LBS (5 years after achieving the LB grade and following the demonstration of performed activity in the 5-year period) grades.

3.4. Training exercises

Practical exercises foreseen during the training include development of the nodalization from the pre-prepared database with problem specifications. To this end, educational material and presentations/lectures on the exercise will be provided with a detailed explanation of the objectives of the work that the trainee must perform. Extensive application of the code by the trainee at his own institution following detailed recommendations and under the supervision of the course lecturers is foreseen as "homework." The use of the code at the course venue is foreseen for the following applications:

(i) fundamental problems including nodalization development;
(ii) basic test facilities and related experiments including nodalization development;
(iii) SETFs and related experiments including nodalization development;
(iv) ITF experiments with nodalization modifications; and
(v) NPP transients including nodalization modifications.

For each of the above cases, the trainee will be required to

(1) develop (or modify) a nodalization starting from the database or problem specifications provided;
(2) run the reference test case;
(3) compare the results of the reference test case with data (experimental data, results of other codes, and analytical solution);
(4) run sensitivity calculations;
(5) produce a comprehensive calculation report following a prescribed format whereby the report should include, for example,
    (a) the description of a particular facility;
    (b) the description of an experiment (including relevance to scaling and relevance to safety);
    (c) modalities for developing (or modifying) the nodalization;
    (d) the description and use of nodalization qualification criteria for steady-state and transient calculations;
    (e) qualitative and quantitative accuracy evaluation;
    (f) use of thresholds for the acceptability of results for the reference case;
    (g) planning and analysis of the sensitivity runs; and
    (h) an overall evaluation of the activity (code capabilities, nodalization adequacy, scaling, impact of the results on the safety and the design of NPP, etc.).

3.5. Examination

On-site examination at different stages during the course is considered a condition for the successful completion of the code user training. The homework that the candidate must complete before attempting the on-site examination includes

(A) studying the material/documents supplied by the course organizers and
(B) solving the problems assigned by the course organizers. This also involves the preparation of suitable reports that must be approved by the course organizers.

The on-site tests consist of four main steps that include the evaluation of the reports prepared by the candidate, answering questions on the reports and course subjects, and demonstrating the capability to work with the selected code. Each step must be accomplished before proceeding to the subsequent one.

4. 3D S.UN.COP SEMINARS: FOLLOW-UP OF THE PROPOSAL

4.1. Background information about 3D S.UN.COP trainings

The 3D S.UN.COP (Scaling, Uncertainty, and 3D coupled code calculations) training aims to transfer competence, knowledge, and experience from recognized international experts in the area of scaling, uncertainty, and 3D coupled code calculations in nuclear reactor safety technology to analysts with a suitable background in nuclear technology.

The training (http://dimnp.ing.unipi.it/3dsuncop) is open to research organizations, companies, vendors, industry, academic institutions, regulatory authorities, national laboratories, and so on. The seminar is in general subdivided into three parts and participants may choose to attend a one-, two-, or three-week course. The first week is dedicated to the background information including the theoretical bases for the proposed methodologies; the second week is devoted to the practical application of the methodologies and to the hands-on training on numerical codes; the third week is dedicated to the user qualification problem through the hands-on training for advanced user and includes a final exam. From the point of view of the conduct of the training, the weeks are characterized by lectures, code-expert teaching, and by hands-on-application. More than thirty scientists are in general involved in the organization of the seminars, presenting theoretical aspects of the proposed methodologies and holding the training and the final examination. A certificate of qualified code user is released to participants

Table 1: Subjects and time schedule necessary for the LA Code user grade.

Code user grade	Weeks	Lectures	Specific for homework	Homework	On-site test
	1-2	A1, A2^, B1, B2^, C1, D1			
	3		C2, D2		
	4–25			A, B, C2*, D2*	
	26				A1, B1, C, D, C2°, D2°
	27	A2, E1	E2		
	28–50			E2*	
LA	51				A2, E, E2°
	52	B2, F1	F2		
	53–76			F2*	
	77				B2, F, F2°
	78	H, G1	G2*		
	79–102			G2*	
	103				G, H, G2*
LB (5 yrs after LA)	1				I*, J, K, K°
LBS (5 yrs after LB)	1				L*

^ Fundamental
* Report necessary
° Solution of submitted problems and discussion.

that successfully solve the assigned problems during the exams.

The framework in which the 3D S.UN.COP seminars have been designed may be derived from Figure 2, where the roles of two main international institutions (OECD and IAEA) and of the US NRC (here playing the role of any other regulatory body of other countries) to address the problem of user effect are outlined together with the proposed programs and produced documents. Figure 3 depicts how the 3D S.UN.COP ensures the nuclear technology maintenance and advancements through the qualification of personnel in regulatory bodies, research activities, and industries by means of teaching by very well-known scientists belonging to the same type of institutions.

Seven training courses have been organized up to now and were successfully held at

(i) The University of Pisa (Pisa, Italy), 5–9 January 2004 (6 participants);

(ii) The Pennsylvania State University (University Park, PA, USA), 24–28 May 2004 (15 participants);

(iii) The University of Pisa (Pisa, Italy), 14–18 June 2004 (11 participants);

(iv) The University of Zagreb (Zagreb, Croatia), 20 June–8 July 2005 (19 participants);

(v) The Technical University of Catalonia (Barcelona, Spain), 23 January–10 February 2006 (33 participants);

(vi) The "Autoridad Regulatoria Nuclear (ARN)," the "Comisión Nacional de Energía Atómica (CNEA)," the "Nucleoelectrica Argentina S.A (NA-SA)," and the "Universidad Argentina De la Empresa" (Buenos Aires, Argentina), 2 October–14 October 2006 (37 participants); and

(vii) The Texas A&M University (College Station, Texas, USA), 22 January–9 February 2007 (26 participants).

4.2. *Objectives and features of the 3D S.UN.COP seminar trainings*

The main objective of the seminar activity is the training in safety analysis of analysts with a suitable background in nuclear technology. The training is devoted to the promotion and use of international guidance and to homogenize the approach to the use of computer codes for accident analysis. The main objectives are

(i) to transfer knowledge and expertise in the Uncertainty Methodologies, Thermal-Hydraulics System Code, and 3D Coupled Code Applications;
(ii) to diffuse the use of international guidance;
(iii) to homogenize the approach in the use of computer codes (like RELAP, TRACE, CATHARE, ATHLET, CATHENA, PARC, RELAP/SCDAP, MELCOR, and IMPACT) for accident analysis;
(iv) to disseminate the use of standard procedures for qualifying thermal-hydraulic system code calculation (e.g., through the application of the UMAE "uncertainty methodology based on accuracy extrapolation" [15]);
(v) to promote best estimate plus uncertainty (BEPU) methodologies in thermal-hydraulic accident analysis through the presentation of the current industrial applications [16–20] and the description of the theoretical aspects of the deterministic and statistical uncertainty methods as well as the method based upon the propagation of output errors (called CIAU "code with the capability of internal assessment of uncertainty" [21, 22]);
(vi) to spread available robust approaches based on BEPU methodology in licensing process;

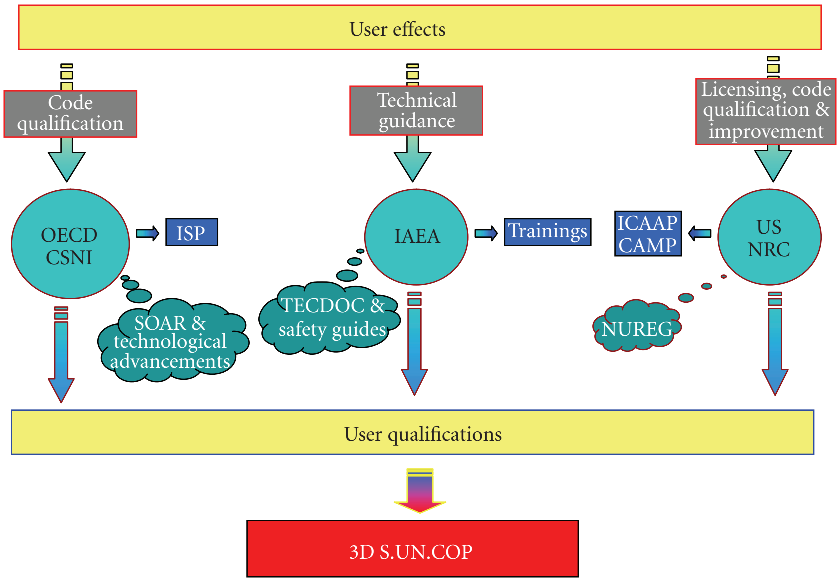

FIGURE 2: 3D S.UN.COP framework to address the user effect problem.

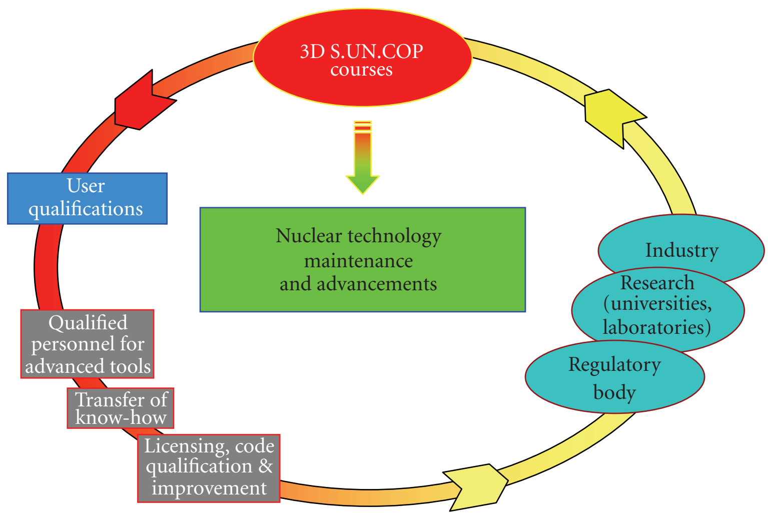

FIGURE 3: 3D S.UN.COP Loop of benefits.

(vii) to address and reduce user effects; and

(viii) to realize a meeting point for exchanges of ideas among the worlds of Academy, Research Laboratories, Industry,Regulatory Authorities, and International Institutions.

The main features of the seminar course are identified as follows.

(i) *The practical use of a mix of different codes*. The use of different code is worthwhile to establish a common basis for code assessment and for the acceptability of code results.

(ii) *The exam*. Exams were in the past courses (very) well accepted by code users. The exam gives them the possibility to show their expertise and to demonstrate the effort done during the course.

(iii) *The practical use of procedures for nodalization qualification*. Standardized techniques for developing and qualifying nodalization (i.e., input) can be directly applied in the participants institutions.

(iv) *The practical use of procedures for accuracy quantification*. The availability of methodologies and tools for quantifying qualitatively and quantitatively the accuracy (i.e., the discrepancy between experimental and calculated data) constitutes a key point for the acceptability of the code results.

(v) *The "joining" between BE codes and uncertainty evaluation*. The use of BEPU methodology within the licensing process is worthwhile for predicting more "realistic" results and for demonstrating the existence of larger safety margins.

(vi) *The large participation of very well-known international experts.* The establishment, integrity, and use of international guidance are promoted through lectures presented by top-level scientists coming from different institutions and countries.

4.3. Scientific and technological areas presented at the 3D S.UN.COP

As the acronym 3D S.UN.COP implies, the following three scientifically relevant areas for the nuclear technology are addressed during the course.

(1) Scaling analysis.
(2) Best estimate plus uncertainty analysis.
(3) Three-dimensional coupled code analysis.

Brief descriptions of each topic are given hereafter.

4.3.1. Scaling analysis

Scaling is a broad term used in nuclear reactor technology, as well as in basic fluid dynamics, and in thermal hydraulics. 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 as well as adopted materials, including working fluids, and different ranges of variation for thermal-hydraulic quantities.

Therefore, the word "scaling" may have different meanings in different contexts. In system thermal hydraulics, a scaling process, based upon suitable physical principles, aims at establishing a correlation between (a) phenomena expected in a NPP transient scenario and phenomena measured in smaller scale facilities or (b) phenomena predicted by numerical tools qualified against experiments performed in small scale facilities (in connection with this point, 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).

Three main objectives can be associated to the scaling analysis:

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

In order to address the scaling issue, different approaches have been historically followed:

(i) fluid balance equation, deriving nondimensional parameters adopting the Buckingham theorem,
(ii) semi-empirical mechanistic equations, deriving non-dimensional parameters,
(iii) to perform experiments at different scales (very expensive way and could not be totally exhaustive),
(iv) to develop, to qualify, and to apply codes showing their capabilities at different scales.

The first item recalls a typical approach based on a theorem (applied also to solve heat transfer problems) for determining the number of independent nondimensional groups needed to describe a phenomenon. It stated that a physical relationship among n variables, which can be expressed in a minimum of m dimensions, can be rearranged into a relationship among $(n - m)$ independent dimensionless groups of the original variables. Buckingham called the dimensionless groups pi-groups and identified them as $\Pi_1, \Pi_2, \Pi_3, \ldots, \Pi_{n-m}$. Thus a dimensional functional equation reduces to a dimensionless functional equation of the form

$$\Pi_1 = f(\Pi_2, \Pi_3, \ldots, \Pi_{n-m}). \tag{1}$$

The second item implies the definition of non-dimensional parameters derived from relationships that link in an empirical way some dependency, for example, from consideration of experimental evidence. Again dimensionless groups are defined similar to the pi-groups. It should be reminded that the relationships defined in this approach are valid for a restricted range thus also the dimensionless parameters are affected by this limitation.

Performing experiment at different scale (third item) might be a way to solve the scaling problem but firstly a lot of experiments should be conducted to cope with the wide range of the scaling factor, secondly the experimental results are affected by peculiarity related to the typical dimension of a test rig at a certain scale.

The last proposal to solve the scaling problem (fourth item) is to accept all the limitation remarked above, to develop a system code, to qualify it against experimental data, to prove that its accuracy is scale independent, and to apply such code to predict the same relevant phenomena that are expected to find in a same experiment (or transient) performed at different scale.

4.3.2. Best-estimate plus uncertainty analysis

In the past, large uncertainties in the computer models used for nuclear power system design and licensing have been compensated using highly conservative assumptions. The loss-of-coolant-accident (LOCA) evaluation model is one of the main examples about this approach. Conservative analysis was introduced to cover uncertainties at the level of knowledge in the 1970s and it is based on the variation of key components of the safety analysis (computer code, availability of components and systems, and initial and boundary conditions) in a way leading to pessimistic results relative to specified acceptance criteria. However, the results obtained by this approach may be misleading (e.g., unrealistic behavior may be predicted or order of events may be changed) and this typically leads to unphysical results. In addition,significant economic penalties, not necessarily commensurate to the safety benefits, may result as consequence of the unknown level of used conservatism. As a conclusion, the use of this approach is not longer recommended (e.g., in [23], however it is still mandatory in the USA for methodologies referencing the Appendix K of US NRC 10 Code of Federal Regulations 50 (10 CFR 50) [24]) and today the application of “realistic” code methods rather than “conservative” approaches can be identified.

By definition, a best estimate (BE) analysis (the term “best estimate” is usually used as a substitute for “realistic”) is an accident analysis which is free of deliberate pessimism regarding selected acceptance criteria, and is characterized by applying best estimate codes along with nominal plant data and with best estimate initial and boundary conditions. However, notwithstanding the important achievements and progress made in recent years, the predictions of the best estimate system codes are not exact but remain uncertain because [7] of the following.

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

Consequently, the results of the code calculations may not be applicable to give exact information on the behavior of a NPP during postulated accident scenarios. Therefore, best estimate predictions of NPP scenarios must be supplemented by proper uncertainty evaluations in order to be meaningful. The term “best estimate plus uncertainty” (BEPU) was coined for indicating an accident analysis which

(1) is free of deliberate pessimism regarding selected acceptance criteria,
(2) uses a BE code, and
(3) includes uncertainty analysis.

Thus the word “uncertainty” and the need for uncertainty evaluation are strictly connected with the use of BE codes and, at least, the following three main reasons for the use of uncertainty analysis can be identified.

(i) *Licensing and safety:* if calculations are performed in best estimate fashion with quantification of uncertainties, a “relaxation” of licensing rules is possible and a more realistic estimates of NPPs’ safety margins can be obtained.
(ii) *Accident management:* the estimate of code uncertainties may also have potential for improvements in emergency response guidelines.
(iii) *Research prioritization:* the uncertainty analysis can help to identify correlation and code models that need the most improvement (code development and validation become more cost effective); it also shows what kind of experimental tests are most needed.

Development of the BEPU approach has spanned nearly the last three decades. The international project on the evaluation of various BEPU methods—uncertainty methods study (UMS)—conducted under the administration of the OECD/NEA [7] during 1995–1998 already concluded that the methods are suitable for use under different circumstances and uncertainty analysis is needed if useful conclusions are to be obtained from best estimate codes. Similar international projects are in progress under the administration of OECD/NEA (BEMUSE—best estimate methods uncertainty and sensitivity evaluation [25]) and IAEA (Coordinated Research Project on investigation of uncertainties in best estimate accident analyses) to evaluate the practicability, quality, and reliability of BEPU methods.

Notwithstanding the above considerations, it is necessary to note that the selection of a BEPU analysis in place of a conservative one depends upon a number of conditions that are away from the analysis itself. These include the available computational tools, the expertise inside the organization, the availability of suitable NPP data (e.g., the amount of data and the related details can be much different in the cases of best estimate or conservative analyses), or the requests from the national regulatory body (e.g., in US licensing process, the BEPU approach was formulated as an alternative to Appendix K conservative approach defined in [24] to reflect the improved understanding of Emergency Core Cooling System (ECCS) performance obtained through the extensive research [1, 26]). In addition, conservative analyses are still widely used to avoid the need of developing realistic models based on experimental data or simply to avoid the burden to change approved code and/or the approaches or procedures to get the licensing.

4.3.3. *Three-dimensional coupled code analysis*

The advent of increased computing power with the present available computer systems is making possible the coupling of large codes that have been developed to meet specific needs such as three-dimensional neutronics calculations for partial anticipated transients without scram (ATWS), with computational fluid dynamics codes, and to study mixing in three-dimensions (particularly for passive emergency core cooling systems) and with other computational tools. The range of software packages that are desirable to couple with advanced thermal-hydraulics systems analysis codes includes

(i) multidimensional neutronics,
(ii) multidimensional computational fluid dynamics (CFD),
(iii) containment,
(iv) structural mechanics,
(v) fuel behavior, and
(vi) 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 is dependent on the phenomena that are being modeled and analyzed. 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 behavior of fluid moving through the core region, where a portion of the core is modeled in great detail using a CFD code while the remainder of the core is modeled using a system analysis code would require tight coupling

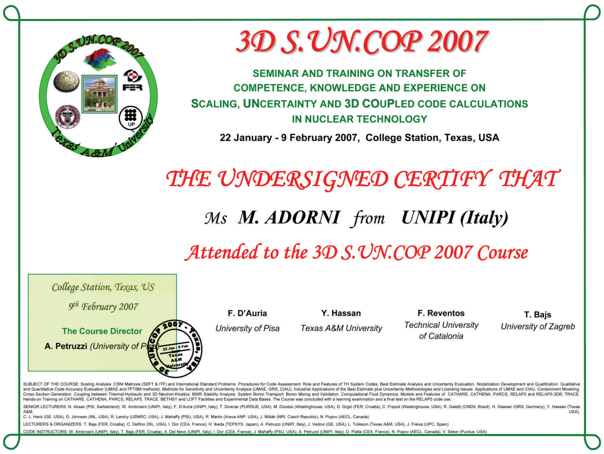

Figure 4: 3D S.UN.COP "LA Code User Grade" Certificate.

if the two codes were linked—since dramatic changes may occur during a NPP transient. Indeed, since CFD codes generally do not have the capability to model general system behavior due to the exceedingly large computer resource requirements, the only means to update a CFD analysis of a somewhat rapid transient in an NPP core region is via close coupling with a system analysis code used to model the NPP system. Thus the system analysis code provides boundary conditions to the CFD code if such an analysis need is identified.

4.4. The structure of the 3D S.UN.COP

The seminar is subdivided into three main parts, each one with a program to be developed in one week.The changes between lectures, computer work, and model discussion have shown to be useful at maintaining participant interest at a high level. The duration of the individual sessions varied substantially according to the complexity of the subjects and the training needs of the participants.

(i) The first week (titled "fundamental theoretical aspects") is fully dedicated to lectures describing the concepts of the proposed methodologies.The following technical sessions (with more than 40 lectures) are presented covering the main topics hereafter listed.

Session I: System codes: evaluation, application, modeling, and scaling

(1) Models and capabilities of system code models,
(2) Development process of generic codes and developmental assessment,
(3) Scaling of thermal-hydraulic phenomena,
(4) Separate and integral test facility matrices.

Session II: International standard problems

(1) Lesson learned from OECD/CSNI ISP,
(2) Characterization and Results from some ISP.

Session III: Best estimate in system code applications and uncertainty evaluation

(1) IAEA safety standards,
(2) Origins of uncertainty,
(3) Approaches to calculate uncertainty,
(4) User effect,
(5) Evaluation of safety margins using BEPU methodologies,
(6) International programs on uncertainty (UMS [7] and BEMUSE [25]).

Session IV: Qualification procedures

(1) Qualifying, validating, and documenting input,
(2) The feature of UMAE methodology,
(3) Description and use of nodalization qualification criteria for steady-state and transient calculations,
(4) Use of thresholds for the acceptability of results for the reference case,
(5) Qualitative accuracy evaluation,
(6) Quantitative accuracy evaluation by fast Fourier transform based method (FFTBM).

Session V: Methods for sensitivity and uncertainty analysis

(1) GRS statistical uncertainty methodology [27],
(2) CIAU method for uncertainty evaluation,
(3) Adjoint sensitivity analysis procedure (ASAP) and global adjoint sensitivity analysis procedure (GASAP), procedures for sensitivity analysis [28, 29],
(4) Comparison of uncertainty methods with code scaling, applicability, and uncertainty (CSAU) evaluation methodology [6].

Session VI: Relevant topics in best estimate licensing approach

(1) Best estimate approach in the licensing process in several countries (e.g., Brazil, Germany, US, etc.).

Session VII: Industrial application of the best estimate plus uncertainty methodology

(1) Westinghouse realistic large break LOCA methodology [16],
(2) AREVA realistic accident analysis methodology [17],
(3) GE technology for establishing and confirming uncertainties [18],
(4) Best estimate and uncertainty BEAU for CANDU reactors [19],
(5) UMAE/CIAU application to Angra-2 licensing calculation [20].

(ii) The second week (titled "Practical Applications and Hands-on Training") is devoted to lectures on the practical aspects of the proposed methodologies and to the hands-on training on numerical codes like ATHLET, CATHARE, CATHENA, RELAP5 USNRC, RELAP5-3D, TRACE, PARCS, RELAP/SCDAP, and IMPACT. The following technical sessions are presented covering the main topics hereafter listed.

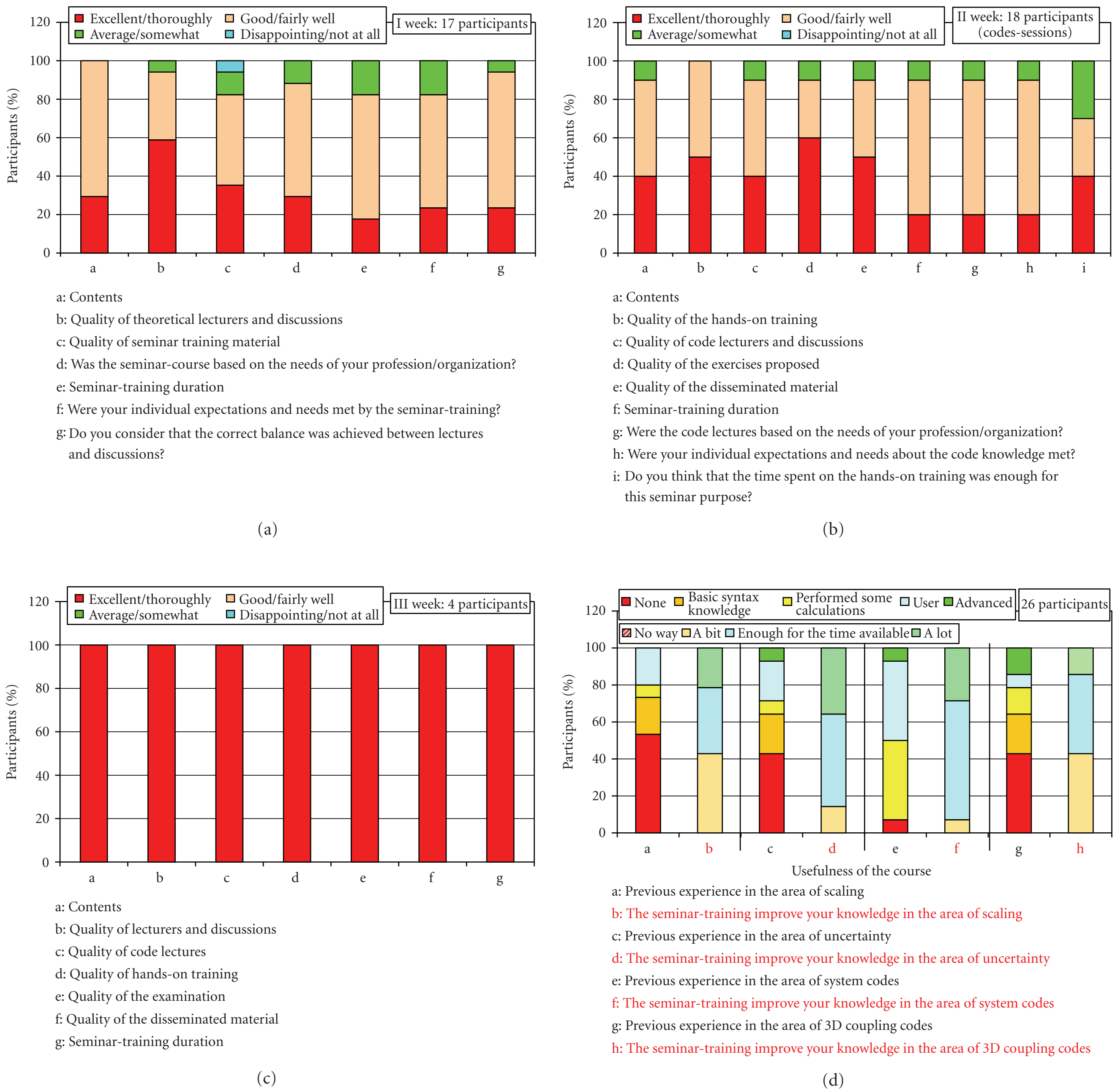

Figure 5: Design and conduct of the seminar training.

Session I: Coupling methodologies

(1) Cross-section generation: models and applications,
(2) Coupling 3D neutron-kinetics/thermal-hydraulic codes (3D NK-TH),
(3) Uncertainties in basic cross-section,
(4) CIAU extension to 3D NK-TH.

Session II: Coupling code applications

(1) PWR-BWR-WWERanalysis,
(2) BWR stabilityissue,
(3) WWER containment modeling,
(4) System boron transport, boron mixing and validation.

Session III: CIAU/UMAE applications

(1) Key applications of CIAU methodology,
(2) Example of code results from application to ITF (LOFT, LOBI, BETHSY) and to a NPP (PWR-Type and WWER-Type),
(3) “PSB Facility” counterpart test,
(4) Bifurcation study with CIAU,
(5) CIAU software.

Table 2: 3D S.UN.COP 2007.

Structure of 3D S.UN.COP 2007		
Week	Content	# Part.
I-Fundamental Theoretical Aspects	7 Technical Session—35 Lectures	19
II-Practical Applications and Hands-on Training—(*lectures sessions*)	3 Technical Session—15 Lectures	20
III-Practical Applications and Hands-on Training—(*parallel code sessions of 20 hours*)	RELAP5 USNRC and RELAP5-3D	5
	TRACE	5
	PARCS	5
	CATHARE	3
	CATHENA	2
IV-Hands-on Training for Advanced Users and Final Examination	24 hours hand-on trainings–2 days exam	4

Participants, lectures, and code developers			
Participants		Lectures and code developers	
Institutions	Country	Names	Institutions, Country
COMENA	Algeria	N. Aksan	PSI, Switzerland
AECL	Canada	W. Ambrosini	UNIPI, Italy
Nuclear Safety Solution		T. Bajs	FER, Croatia
NPIC	China	F. D'Auria	UNIPI, Italy
University of Zagreb	Croatia	C. Delfino	ISL, USA
EDF	France	I. Dor	CEA France
FZK	Germany	T. Downar	PURDUE, USA
ENEA	Italy	M. Dzodzo	Westinghouse, USA
University of Pisa		A. Del Nevo	UNIPI, Italy
PBMR	South Africa	C. Frepoli	Westinghouse, USA
ESKOM		R. Galetti	CNEN, Brazil
KEPRI	South Korea	J. Gary	INL, USA
Technical University of Catalonia	Spain	H. Glaeser	GRS, Germany
		D. Grgic	FER, Croatia
IBERINCO		Y. Hassan	Texas A&M, USA
ROLLS-ROYCE	UK	C. Heck	GE, USA
AREVA NP	USA	H. Ikeda	TEPSYS, Japan
Texas A&M		R. Landry	US NRC, USA
		R. Martin	AREVA, USA
		J. Mahaffy	PSU, USA
		J. Misak	NRI, Czech Republic
		A. Petruzzi	UNIPI, Italy
		D. Pialla	CEA, France
		N. Popov	AECL, Canada
		C. Pretel	ETSEIB, Spain
		F. Reventos	ETSEIB, Spain
		J. Vedovi	GE, USA
		S. Volkan	PURDUE, USA

Session IV: Computational Fluid Dynamics Codes

(1) The role and the structure of the computational fluid dynamics (CFD) codes,
(2) CFD simulation in nuclear application: needs and applications.

Each of the *parallel hands-on trainings on numerical codes* consists of about 20 hours and covers the following main topics:

(1) structure of specific codes,
(2) numerical methods,

(3) description of input decks,
(4) description of fundamental analytical problems,
(5) analysis and code hands-on training on fundamental problems (e.g., for RELAP5, fundamental proposed problems deal with boiling channel, blow-down of a pressurized vessel, and pressurizer behavior),
(6) Example of code results from applications to ITFs (LOFT, LOBI, BETHSY).

(iii) The third week (titled "Hands-on Training for Advanced Users and Final Examination") is designed for advanced users addressing the user effect problem. The participants are divided into groups of three and each group receives the training from one teacher. The applications of the proposed methodologies (UMAE, CIAU, etc.) are illustrated through the BETHSY ISP 27 (small break LOCA) and LOFT L2–5 (large break LOCA) tests. Applications and exercises using several tools (RELAP5, WinGraf, FFTBM, UBEP, CIAU, etc.) are considered. The following main topics are covered:

(1) modalities for developing (or modifying) the nodalization,
(2) plant accident and transient analyses,
(3) examples of code results from application to a NPP (PWR-Type and VVER-Type), and
(4) Code hands-on training through the application of system codes to ITFs (LOFT and BETHSY).

A final examination on the lessons learned during the seminar is designed and consists of three parts.

(i) Written Part: questions about the topics discussed during the seminar are proposed and assigned both to each participant and to each group.
(ii) Application Part: two types of problems are proposed to the single participant and to the group, respectively.
    (1) *Detection of Simple Input Error:*
    Each participant receives the experimental data of the selected transient, the correct RELAP5 nodalization input deck, and the restart file of the wrong input deck containing one simple input error. Each participant will identify the error.
    (2) *Detection of Complex Input Error:*
    Each group receives the experimental data of the selected transient, the correct RELAP5 nodalization input deck, and the restart file of the wrong input deck containing one complex input error. Each group will identify the error.

    Evaluation reports are submitted in a written form containing short notes about the reasons for the differences between results of the reference calculation and results from the "modified" nodalization. At least, one problem over two will be correctly solved to obtain the certificate.
(iii) Final Discussion: each participant takes an oral examination discussing own results (or results obtained by own group) with the examiners. General questions related to lectures presented during the three-week seminar are asked to the participants.

A certificate of type "LA Code User Grade" (see Table 1) like the one depicted in Figure 4 is released to participants that successfully solved the assigned problems.

4.5. 3D S.UN.COP 2007 at Texas A&M University (Texas, USA)

The 3D S.UN.COP 2007 was successfully held at the Texas A&M University (Texas, USA) from January 22nd to February 9th with the attendance of 26 participants coming from 12 countries and 17 different institutions (universities, vendors, national laboratories, and regulatory bodies). About 30 scientists (from 11 countries and 19 different institutions) were involved in the organization of the seminar, presenting theoretical aspects of the proposed methodologies and holding the training and the final examination. More details may be found in Table 2.

All the participants achieved a basic capability to set up, run, and evaluate the results of a thermal-hydraulic system code (e.g., RELAP5) through the application of the proposed qualitative and quantitative accuracy evaluation procedures.

At the endof the seminar a questionnaire for the evaluation of the course was distributed to the participants. All of them very positively evaluated the conduct of the training as can be derived from Figure 5.

5. CONCLUSIONS

An effort is being made to develop a proposal for a systematic approach to user training. The estimated duration of training at the course venue, including a set of training seminars, workshops, and practical exercises, is approximately two years. In addition, the specification and assignment of tasks to be performed by the participants at their home institutions, with continuous supervision from the training center, have been foreseen.

The 3D S.UN.COP seminars training courses constitute the follow-up of the presented proposal. The problem of the code-user effect along with the methodologies for performing the scaling-, the BEPU-, and the 3D coupled-code-calculation-analyses are the main topics discussed during the course. The responses of the participants during the training demonstrated an increase in their capabilities to develop and/or modify the nodalizations and to perform a qualitative and quantitative accuracy evaluation. It is expected that the participants will be able to set up more accurate, reliable, and efficient simulation models applying the procedures for qualifying the thermal-hydraulic system code calculations and for the evaluation of theuncertainty.

LIST OF ABBREVIATIONS

ASAP:	Adjoint sensitivity analysis procedure
ATWS:	Anticipated transients without scram
BE:	Best estimate
BEAU:	Best estimate and uncertainty
BEMUSE:	Best estimate methods uncertainty and sensitivity evaluation
BEPU:	Best estimate plus uncertainty

BETF:	Basic experiments test facilities
BoP:	Balance of plant
BWR:	Boiling water reactor
CFD:	Computational fluid dynamics
CFR:	Code of federal regulations
CIAU:	Code with the capability of Internal Assessment of Uncertainty
CSAU:	Code scaling, applicability and uncertainty evaluation
CSNI:	Committee on the Safety of Nuclear Installations
ECCS:	Emergency core cooling system
EVET:	Equal velocities, equal temperatures
FFTBM:	Fast fourier transform-based method
FP:	Fundamental problem
GASAP:	Global adjoint sensitivity analysis procedure
HEM:	Homogeneous equilibrium model
IAEA:	International Atomic Energy Agency
ISP:	International standard problems
ITF:	Integral test facilities
LA:	Level A degree (terminology used in the certificate)
LB:	Level B degree (terminology used in the certificate)
LBS:	Level B Senior degree (terminology used in the certificate)
LOCA:	Loss-of-coolant-accident
NEA:	Nuclear Energy Agency
NK:	Neutron-kinetics
NPP:	Nuclear power plants
OECD:	Organization for Economic Cooperation and Development
PWR:	Pressurized water reactor
SETF:	Separate effect test facility
TH:	Thermal-Hydraulic
UBEP:	Uncertainty band extrapolation process
UMAE:	Uncertainty methodology based on acuracy extrapolation
UMS:	Uncertainty methods study
US NRC:	United States Nuclear Regulatory Commission
UVUT(UP):	Unequal velocities, unequal temperatures (unequal pressure)
WWER:	Water-cooled water-moderated energy reactor
1D, 3D:	One-dimensional, three-dimensional
3D S.UN.COP:	(Training on) Scaling, Uncertainty, and 3D coupled code calculations.

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Review Article

International Standard Problems and Small Break Loss-of-Coolant Accident (SBLOCA)

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Recommended by Cesare Frepoli

Best-estimate thermal-hydraulic system codes are widely used to perform safety and licensing analyses of nuclear power plants and also used in the design of advance reactors. Evaluation of the capabilities and the performance of these codes can be accomplished by comparing the code predictions with measured experimental data obtained on different test facilities. OECD/NEA Committee on the Safety of Nuclear Installations (CSNI) has promoted, over the last twenty-nine years, some forty-eight international standard problems (ISPs). These ISPs were performed in different fields as in-vessel thermal-hydraulic behaviour, fuel behaviour under accident conditions, fission product release and transport, core/concrete interactions, hydrogen distribution and mixing, containment thermal-hydraulic behaviour. 80% of these ISPs were related to the working domain of principal working group no.2 on coolant system behaviour (PWG2) and were one of the major PWG2 activities for many years. A global review and synthesis on the contribution that ISPs have made to address nuclear reactor safety issues was initiated by CSNI-PWG2 and an overview on the subject of small break LOCA ISPs is given in this paper based on a report prepared by a writing group. In addition, the relevance of small break LOCA in a PWR with relation to nuclear reactor safety and the reorientation of the reactor safety program after TMI-2 accident are shortly summarized. The experiments in four integral test facilities, LOBI, SPES, BETHSY, ROSA IV/LSTF and the recorded data during a steam generator tube rupture transient in the DOEL-2 PWR (Belgium) were the basis of the five small break LOCA related ISP exercises, which deal with the phenomenon typical of small break LOCAs in Western design PWRs. Some lessons learned from these small break LOCA ISPs are identified in relation to code deficiencies and capabilities, progress in the code capabilities, possibility of scaling, and various additional aspects. ISPs are providing unique material and benefits for some safety-related issues.

1. INTRODUCTION

Large transient thermal-hydraulic system codes are widely used to perform safety and licensing analyses of nuclear power plants and also used in the design of advanced reactors. Evaluation of the capabilities and the performance of these codes can be accomplished by comparing the code predictions with measured experimental data obtained on different test facilities. In this respect, parallel to other national and international programmes, OECD Nuclear Energy Agency (OECD/NEA) Committee on the Safety of Nuclear Installations (CSNI) has promoted, over the last thirty years some fourty eight international standard problems (ISPs) [1, 2]. The first international standard problem (ISP) was organized in 1975 on the famous "Edwards blowdown pipe" experiment. These ISPs were performed in different fields as in-vessel thermal-hydraulic behaviour, fuel behaviour under accident conditions, fission product release and transport, core/concrete interactions, hydrogen distribution and mixing, and containment thermal-hydraulics. Roughly, 60% of these ISPs concerned the thermal-hydraulic behaviour.

The main goal of ISP exercises is to increase confidence in the validity and the use of the different tools that are used in assessing the safety of nuclear installations. These tools may vary to some extent in different countries and are extremely complex. Therefore, the ISPs were considered as an effective way to get a common understanding and judgment about the code/user capabilities on an international basis. Indeed, in an ISP the predictions of different computer codes with respect to a given physical problem may be compared with the results of an experiment or/and among each other.

While the developmental assessment still belongs to the organisation developing the codes, ISP exercises can be considered as a complementary activity, assessing the codes through the analysis of experts different from the code developers and covering much wider ranges, specifically in terms of thermal-hydraulics scenarios and value of parameters.

The objectives of the ISP may be summarized as

(i) to contribute to better understanding of postulated events,
(ii) to compare and evaluate the capability of codes (mainly best estimate codes),
(iii) to suggest improvements to the code developers,
(iv) to improve the ability of code users,
(v) to address the so called scaling effect.

Standard problems are performed as "open" or "blind" (double blind) problems. In an "open" problem, all participants know the results of the experiment in detail before performing their calculations. In a "blind" exercise, the results are locked until the code users submit the calculation results for comparisons. A so called "double blind" exercise consists of a "blind" one for which no other experimental data related to the test facility has been published or made available to the ISP participants before submission of results. For blind exercises the participants are keenly encouraged to run post test calculations when the experimental results are released. Those post test calculations are sensitivity studies, where various options and/or models are tested in order to see how they affect the results, also to better understand the reasons for eventual discrepancies resulting from comparing "blind" results and experimental data.

As mentioned in [3], both integral and separate effect experiments may be considered for ISP exercise. Also best-estimate codes are preferably used. The reader will also find in the same reference a complete description of the organisation of an ISP exercise.

A global review and synthesis on the contribution that small break LOCA ISPs have made to address nuclear reactor safety issues was initiated by the principal working group no. 2 (PWG2) in September 1993. Further to this request of the PWG2, an action has been put, during the thirteenth meeting of the Task Group on Thermal-Hydraulic System Behaviour (TG-THSB), to carry out this review and synthesis work on previous small break LOCA ISPs. As a result of this synthesis work, a short overview report was written on this subject [4] by a group of experts in the TG-THSB. In order to limit the effort, five ISPs were selected for this evaluation, but not strictly based on small break LOCA scenarios; ISPs in which similar phenomenon to small break LOCA was observed are also considered

(i) *ISP 18:* LOBI Mod2 1% small break LOCA [5];
(ii) *ISP 20:* Doel 2 steam generator tube rupture event [6];
(iii) *ISP 22:* SPES-simulating loss of feedwater transient in Italian PWR [7, 8];
(iv) *ISP 26:* ROSA-IV LSTF 5% cold leg small break LOCA experiment [9];
(v) *ISP 27:* BETHSY 0.5% small break LOCA with loss of high-pressure injection [10].

The ISPs 18, 22, and 27 were "blind" exercises, while the ISPs 20 and 26 were "open" ones. The ISP 18 is the "oldest" ISP retained in this review and synthesis work, since such an ISP may be considered as a milestone in the transition process between the first generation codes (i.e., RELAP4) and the new generation of advanced computer codes (e.g., TRAC, RELAP5, ATHLET, CATHARE). It is to be noted that there were small break LOCA ISP exercises previous to ISP-18, for example, LOFT and semiscale small break LOCA tests, but they were not considered in this review process due to advancement of the codes relative to the application of the first generation codes in these ISPs. Moreover, at that time some of these new codes were in their development phase. In addition, one may consider that, since 1985, the objectives of ISP were slightly changed due to the reason that all codes passed their developmental phase.

While the ISP 22 initiating event is not a small break, it has been considered in this evaluation since specific phenomena observed during the experiment are similar to those observed during small break accident. Moreover, it might give the opportunity to fill the gap between BETHSY and LOBI test facilities for scaling purposes.

ISP 20 has been retained in this evaluation as far as scaling effect has to be addressed. Indeed, the ISP 20 is the unique exercise based on a transient occurring in a full-scale two-loop PWR nuclear plant.

Other internationally conducted research programmes in this same area have been completed in the time period here considered, including ISPs, for example, ISP 25 and ISP 33. Examples are the OECD-LOFT project or LOBI experiments analyzed by a CEC devoted task group. However, resources limitations and willingness to keep some homogeneity for the discussed transients (i.e., ISP 25 is based on a separate effects test, ISP 33 addressed the behaviour of WWER plants; LOFT is a nuclear facility scaled down with criteria different from those of LOBI, SPES, BETHSY, and LSTF; in addition most of the LOFT, LOBI, and LSTF data were not openly available to the whole OECD community) supported the conclusion to restrict the investigation range, though recognizing the fundamental contributions given by the above mentioned programmes in this same area.

The outcome from each considered ISP and in particular the evaluation of the comparisons between measured and predicted system behaviours are described in detail in the "final comparison reports," from [5] to [10], and therefore will not be repeated here. Identically, this synthesis work will not deal with the "user effects" that has been separately addressed and analyzed in detail in [11].

In this paper, some of the aspects addressed in [4] will be summarized in order to provide an overview on the lessons learned from the small break LOCA ISPs. Section 2 will give an overview on the development of small break LOCA issue. Main phenomena and relevance of small break LOCA to reactor safety in a PWR are shortly described in Section 3. A short overview of ISPs and expected technical findings are dealt within Sections 4 and 5. After a presentation of the involved facilities and plant and a description of the different selected tests (Section 6); Section 7 deals with relevant ISP statistics. Section 8 presents the "lessons learned" from the selected ISP activities with some conclusions and recommendations. This also constitutes the main objective of the presented activity.

2. ORIGIN OF SMALL BREAK LOCA ISSUE (SYSTEM THERMAL-HYDRAULICS BEFORE AND AFTER TMI-2)

In early 1970s, former US Atomic Energy Commission convened a public hearing to explore the safety question in relation to the effectiveness of systems to mitigate the consequences of a loss of coolant accident in a nuclear reactor, in case it happens. Ultimately, after extensive public hearings, in 1974, the interim regulations were modified to provide a set of specific requirements for computer codes for ECCS analyses in and a new section, 10 CFR 50.46 [12, Appendix K], requiring ECCS meet established standards. This included a definition that LOCAs are hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary up to and including break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. The safety criteria prescribed in 10 CFR 50.46 are applicable to both large and small break LOCAs. That is to say the limits on peak cladding temperature, cladding oxidation, and hydrogen generation must not be exceeded in a design basis accident. Calculations of ECCS performance using the conservative prescriptions of [12, Appendix K] resulted in the large break LOCA generally being the most limiting accident. At the time, there was a major safety research programme to support code development for large break LOCA and also some limited work on small break LOCA.

The March 1979 accident at the Three Mile Island Unit 2 (TMI-2) reactor led to an extensive reorientation of light water reactor safety research programmes and also regulatory changes. The TMI-2 accident was a small break LOCA, an event given significantly less attention because of the major emphasize on the large break LOCA at the time. Consequent to TMI-2, small break LOCA and plant operational transients received major attention. The experimental simulation of the natural circulation phenomena in the primary loops, including those in the two-phase stratified and counter-current flow regimes, is of primary importance to the thermal-hydraulic response of a nuclear power plant during such transients. Since these phenomena are significantly dependent on facility scale and geometry, large-scale tests for a primary system geometry representative of operational nuclear power plants are required. Either operational facilities were modified to carry out small break LOCA experiments or there were new facilities designed and constructed (see Section 4). It is to be noted that unlike the large break LOCA, the sequence of events following a small break LOCA can evolve in a variety of ways. Operator actions, reactor design, ECCS set points, break size, and location will have a bearing how the small break LOCA scenario unfolds. Therefore, in order to predict the integral system behaviour during a small break LOCA, a best-estimate code must have sufficient modelling capabilities to take these factors into account. These codes are also needed to be assessed against integral system tests. After having been successfully assessed against data from a large number of scaled test facilities, best-estimate codes become the ultimate repository of all previous thermal-hydraulic safety research. ISP activities are a part of this process (see Section 4).

3. SMALL BREAK LOCA IN A PWR WITH RELEVANCE TO NUCLEAR REACTOR SAFETY AND MAIN PHENOMENA

The major characteristic difference between a small break and a large break LOCA is in the rates of coolant discharge and pressure variations with time. In general, small break LOCAs are characterized by an extended period (this can be tens of minutes to several hours at the lower end of the break spectrum) after the occurrence of the break, during which the primary system remains at a relatively high pressure and the core remains covered. As soon as the pumps are tripped, either automatically or manually, gravity-controlled phase separation occurs and gravitational forces dominate the flow and distribution of coolant inside the primary system. The subsequent sequence of events, whether or not the core uncovers and is recovered or reflooded, depends not only on the location, shape, and size of the break, but also on the overall behaviour of the primary and secondary systems. This behaviour is strongly influenced by both automatic and operator initiated mitigation measures. In general, the reactor system response to a small break is slower compared to events after a large break. This allows more time, and different possibilities, for operator interventions. Another principal difference is the domination of gravity effects in small breaks versus inertial effects in the large breaks.

It is to be noted that there is no unique path of development of events following a small break LOCA in PWRs. The scenarios may change drastically by many factors such as the reactor design (e.g., U-tube or once-through steam generators, such as TMI-2), the break size, the core bypass size (allowing some fraction of the inlet cold leg flow directly into the core upper structure without passing through the core), and most importantly, by different operator interactions. As an example, the primary circulation pumps may be shut down early in a small break LOCA transient or they may be allowed to run and circulate the coolant through the core for a long time. These alternative actions can make a large difference in the nature of discharge flow, early heat removal from the core, and the liquid inventory in the system after one hour or so in the transient. Another important possibility of different interactions is through the steam generators. The secondary side of steam generators can be isolated (no feed water flow) or they can be used for a controlled heat removal. It is also possible to cool the reactor through the so-called “feed and bleed” process (on the primary side). Either of these actions will have a major effect on the course of the transient. It is not the intent in this section to provide a catalogue of all possible scenarios following small break LOCA accidents. But it is important to note that an adequate set of modelling capabilities for any of the plausible scenarios will be equally adequate for all other relevant scenarios. This is because the phenomena and processes are the same but their interactions and timing of various developments change in different operations. Therefore, in order to predict the integral system behaviour during a small break LOCA, a best-estimate code must have sufficient modelling capabilities to take these factors into account.

During a PWR small break LOCA, there is the potential for three distinct core heat ups. The first heat up is caused by loop seal formation and the manometric core liquid level depression. Naturally occurring events including loop seal clearing and break uncovery mitigate this heat up. The second heat up occurs following the core quench caused by loop seal clearing and is caused by a simple core boiloff. During this period the primary pressure is decreasing to the accumulator set point and the steam produced by the core boiloff leaves the system via the break. Any heat ups that occur during this period are mitigated by the reflood from the accumulator water. The third possible heat up can occur following depletion of the accumulator tanks and before LPIS injection begins. One drawback to the reflood process accompanying the accumulator injection is a decrease in the ongoing depressurisation process such that another possible heat up occurs before the LPIS primary pressure set points are reached and long-term cooling is provided. Various factors affect the magnitudes of the three potential core heat ups. Some examples are break size, break direction and location, availability of HPIS, and the degree of upper head to downcomer bypass flow. Although the magnitudes of the core heat ups may vary, ECCS performance must be such that the criteria, for example, 10 CFR 50.46 [12] is not exceeded.

The interested readers can obtain further details on small break LOCA in [13].

4. A SHORT OVERVIEW OF ISPS AND TECHNICAL DOMAINS COVERED BY THEM

A compilation of all ISPs performed between 1975 and 1997 can be found with a brief description of each ISP in [1] and an extended list of ISPs (from 1975 to 2007) is also provided in Table 1.

The very first ISPs from 1975 to roughly 1980 focused on LOCA thermal-hydraulics as it was one of the main concerns of that time. We find there ISPs based on separate effects tests (Edwards blowdown pipe, CISE blowdown test, Battelle blowdown test, tube reflooding test ERSEC) and ISPs based on the two only available system experiments for PWRs at that time, that is, SEMISCALE and LOFT.

After Three Mile Island (TMI-2) accident, ISPs started to move from the large breaks to the small breaks. They included ISPs on LOFT L3 small break LOCA series tests for PWRs, ROSA III, and FIX II tests for BWRs. Some large break tests were still selected: PKL reflooding test, as reflooding was considered as a remaining issue; LOFT L2-5, as it was a significant "concluding" nuclear test for large breaks.

During this period (beginning 80s), two ISPs were initiated in a new domain for ISPs at that time which was the domain of thermo-mechanical fuel behaviour during LOCA. These were ISPs on REBEKA test (nonnuclear) and on PHEBUS LOCA test (nuclear).

In parallel to the ISPs dealing with the primary circuit, ISPs (in a first step called CASPs) were organized in the beginning of the 80s on containment experiments either system experiments (BATTELLE Model Containment) or very small scale experiment (AAEC-Australia). These ISPs covered large break situations. They were followed in the mid 80s by ISPs on HDR containment tests (large break in PWR) and Marviken test (BWR).

During the second half of the 80s and during the beginning of the 90s, the ISPs related to thermal-hydraulics were characterized by a full and coherent series based on the experiments which were decided and built after TMI in order to well study small break and transient situations including operator actions. They included ISPs on LOBI-mod2, SPES, ROSA IV, BETHSY facilities for PWRs (lessons learned from these ISPs are provided in [4], summary of which is included in this paper), and PIPER-ONE facility for BWRs. Besides this series, one ISP investigating the effect of noncondensable gases on reflood was performed (ACHILLES), and the first and only one ISP based on real plant was organized in 1988 on the DOEL 2 steam generator tube rupture event.

End of the 80s, the interest of ISPs moved clearly to the severe accident area. ISPs on core degradation were held based on CORA (nonnuclear) and PHEBUS SFD (nuclear). Core concrete interaction was investigated with two ISPs (SURC4 and BETA2). Containment questions and especially hydrogen problems were the subject of two ISPs based on HDR and one ISP based on NUPEC test. In addition, an ISP was also organized on FALCON facility to investigate fission product behaviour with simulants.

One of the extensions of domain covered by ISPs is constituted by the move towards VVER related problems with PACTEL ISP (thermal-hydraulics) and CORA VVER ISP (Core degradation).

In continuation of ISPs on thermal-hydraulics and severe accident, shut down states are investigated with an ISP on BETHSY and steam explosions with an ISP on FARO. STORM and RTF experiments provided data for aerosol behaviour in primary circuit and iodine behaviour in containment under severe accident conditions. UMCP facility was used to assess boron dilution models.

Recent ISPs are PANDA test with six different phases related to passive safety systems for advanced light water reactors; QUENCH-06 and PHEBUS FP-1 tests for severe core degradation; and TOSQAN, MISTRA, and ThAI facilities for containment thermal-hydraulics.

This overview shows the extraordinary large range of technical domains, which have been covered by ISPs. These domains reflect of course the successive changes in the area of concern for nuclear reactor safety research. This demonstrates also that the concept of ISP initiated in the thermal-hydraulic area and extended to several other technical areas, is certainly very productive and useful. We will, in the next sections, analyse in general and also for a specific subject of small break LOCA what are the outcomes and the benefits produced by this activity and how it may explain its success.

Table 1: List of CSNI international standard problems (ISPs) [2].

No.	Completion date	Title
1	1975	Standard problem 1-Edwards pipe blowdown test
2	1975	Analysis of semiscale blowdown test 11, LB LOCA
3	1977	CSNI standard problem 3; comparison of LOCA analysis codes, CISE, blowdown
4	1978	United states standard problem 6 and international standard problem 4: comparison of the standard problem calculations with measured experimental data for semiscale test S-02-6, SB LOCA
5	1979	United states standard problem 7 and international standard problem 5: final comparison report on LOFT test L1-4, LB LOCA
6	1978	ISP-6: calculations comparison report-determination of water level and phase separation effects during the initial blowdown phase
7	1979	comparison report on OECD-CSNI LOCA standard problem no. 7: analysis of a reflooding experiment, ERSEC
8	1979	Semiscale MOD1 test S-06-03 (LOFT counterpart test), LB LOCA
9	**1981**	**LOFT test L3-1 preliminary comparison report, SB LOCA**
10	1981	comparison report on OECD-CSNI LOCA standard problem no. 10: "refill and reflood experiment in a simulated PWR primary system (PKL)
11	**1984**	**LOFT L3-5 and L3-6 comparison reports, SB LOCA**
12	**1982**	**ROSA-III 5% small break test, Run 912, BWR-SB LOCA**
13	1983	international standard problem 13 (LOFT experiment L2-5) preliminary comparison report, LB LOCA
14	1985	behaviour of a fuel bundle simulator during a specified heatup and flooding period (REBEKA experiment) (results of posttest analyses)
15	1983	LOCA experiment at FIX-II facility, BWR
16	1985	rupture of a steam line within the hdr containment leading to an early two-phase flow: results of posttest analyses: final comparison report
17	1984	Marviken BWR standard problem
18	**1987**	**LOBI-MOD2 small break LOCA experiment A2-81**
19	1987	behaviour of a fuel rod bundle during a large break LOCA transient with a two-peaks temperature history (PHEBUS Experiment): final comparison report
20	1988	Doel 2 steam generator tube rupture event: final report
21	**1989**	**PIPER-ONE experiment PO-SB-7: simulation of small and intermediate break LOCA for BWRs**
22	1990	SPES-loss of feedwater transient in Italian PWR. final comparison report (1990) and evaluation of posttest analyses (1992).

Table 1: Continued.

No.	Completion date	Title
23	1989	Rupture of a large diameter pipe in the HDR containment
24	1989	SURC-4-core-concrete interaction test
25	1991	ACHILLES-N2 injection from accumulators and faster (best estimate) reflood rates
26	**1992**	**ROSA-IV LSTF-cold-leg small-break LOCA experiment**
27	**1992**	**BETHSY-small break LOCA with Loss of HP injection**
28	1992	PHEBUS SFD B9+-experiment on the degradation of a PWR type core
29	1993	HDR experiment E11.2-hydrogen distribution inside the HDR containment under severe accident conditions: final comparison report
30	1992	BETA II core-concrete interaction experiment (Test V5.1): comparison report
31	1993	CORA-13 experiment on severe fuel damage
32	—	FLHT-6 experiment, cancelled
33	1994	PACTEL-VVER-440 natural circulation stepwise coolant inventory reduction
34	1994	falcon experiments FAL-ISP-1 and FAL-ISP-2, fission product transport
35	1994	NUPEC hydrogen mixing and distribution test M-7-1: final comparison report
36	1996	CORA-VVER severe fuel damage experiment (test W2)
37	1996	VANAM M3-a multi compartment aerosol depletion test with hygroscopic aerosol material-comparison report
38	1997	loss of the residual heat removal system during mid-loop operation (BETHSY)
39	1997	fuel coolant interaction and quenching (FARO)
40	1999	STORM test SR11-aerosol deposition and resuspension in the primary circuit
41	1999	RTF experiment on iodine behaviour in containment under severe accident conditions
42	2003	PANDA tests (six different phases) related to passive safety systems for advanced light water reactors
43	2001	UMCP boron dilution test
44	2002	Four open and one blind KAEVER aerosol depletion tests with three differently soluble materials and uniform thermal-hydraulic conditions with slight volume condensation
45	2003	QUENCH-06, fuel rod bundle behaviour up to and during reflood/quench (severe core damage)
46	2004	PHEBUS in reactor experiment (FP-1) on the degradation, fission product release, circuit and containment behaviour following overheating of an irradiated fuel rod bundle
47	2005	Based on experiments performed in the TOSQAN, MISTRA and ThAI facilities for containment thermal-hydraulics (e.g., gas distribution, natural convection, heat and mass distribution...)

Table 1: Continued.

No.	Completion date	Title
48	2005	Containment capacity (integrity and ageing of components and structures). 1 : 4 scale model of a prestressed concrete containment vessel (PCCV) of a nuclear power plant (SANDIA II mock-up)
	containment analysis standard problems (CASPs)	
CASP-1	1980	Comparison report on OECD-CSNI containment standard problem no. 1: "steamline rupture within a chain of compartments" (Battelle Institute test D15)
CASP-2	1982	Comparison report on OECD-CSNI containment standard problem no. 2: "water line rupture in a branched compartment chain" (Battelle Institute test D16)
CASP-3	1983	Final comparison report for containment standard problem exercise 3 (australian lucas heights blowdown/containment rig, small-scale two-compartments basic containment experiment)

5. THE EXPECTED TECHNICAL FINDINGS FROM ISP ACTIVITY

The basic material of the technical findings from ISP activity is made of the several predictions obtained with several codes by several code users of a given physical experiment. From these material different cross-comparisons can be made which we will now review.

(i) The first class of comparisons is the comparisons between code predictions and experimental results. Such comparisons are evidently contributing to the code assessment. However, some particularities to this contribution should be emphasized.

(a) This assessment belongs of course to the "independent" assessment. Considering the generally very large number and very large variety of participants to ISPs, the "independent" character is certainly one of the most accentuated that we can afford. For those who are thinking that the independence of assessment is a very important feature, the results of ISPs are unique.

(b) The number of code calculations in the comparison between code predictions and experimental results is certainly the largest that we can imagine on a single test. Almost no individual can do such work at least because of financial limitations. Besides this number of calculations, there are numerous differences in the physical models used in the different codes. The comparisons with experimental results are then very instructive on the effect of these models differences on the capabilities to predict the experiment. Often all codes available in OECD countries (and sometimes in the world) are represented during the ISP execution. A complete international view is then obtained on the status of the predictive capabilities of the phenomena studied in the ISP.

(c) It is clear that the large amount of work produced by the participants and by the organizing country requires that no mistake should be done in the process. As a consequence, the experimental test must be first very carefully selected. Therefore, it is very often one of the best and one of the most significant tests of the experimental programme to which it belongs. The organisation of the ISP requires also that all necessary information be transmitted to the participants in a very comprehensive way. Consequently, the organizing country must do a very high control of test results and of documentation. This last requirement led particularly the OECD/NEA working groups to define standards for test documentation. These standards are summarized in the CSNI report no. 17 [3] and have shown to be quite general and useful, in particular, as they have been used in several other areas than ISP. As the need arises, certain revisions are introduced into this report. Finally, the efforts made on the test selection, on the test control and on the test documentation provide most often a technical quality of very high level to the ISPs activities.

(d) The high-level grade of documentation obtained by following the prescribed standards and the strict selection of the tests based on their physical and safety significance make the ISPs tests very good candidates for inclusion in validation matrices. ISPs tests may often be considered as international reference tests. Their already wide distribution and their consequent availability is also a favouring factor for such choices.

(ii) The second class of comparisons is constituted by the comparisons between different codes. It is the common experience of analysts that understanding and analysing the code responses is a very difficult exercise. Indications are most often required in order to give directions for the analyst in

Table 2: Relevant hardware characteristics of considered PWR simulators and Doel-2 nuclear plant.

	Quantity	SPES	LOBI/Mod2	BETHSY	LSTF/ROSA IV	DOEL-2
1	Reference reactor	W-PWR	KWU-PWR	FRA-PWR	W-PWR	W-PWR
	Reference reactor power (MWt)	2775	3900	2700	3423	1187
2	Mximum power (MWt)/% of nominal power	9.0/138	5.4/100	3.0/10	10.0/14	1187.0/100
3	Reported K_v	1/427	1/712	1/100	1/48	1/1
4	No. of rods	97	64	428	1064	21659
5	Operating pressure of primary loop (MPa)	15.	15.8	15.51	15.5	15.5
6	Operating pressure of secondary loop (MPa)	6.1	6.54	6.91	7.3/7.4	5.88
7	Primary loop volume (m^3)	0.622	0.643	2.88	8.3	168.5
8	No. of U-tubes for each steam generators	13/13/13	8/24	34/34/34	141/141	3260/3260
9	Internal diameter of U-tubes (mm)	15.4	19.6	19.7	19.6	19.6
10	L/D ratio of Hot leg	57.2	73.1/119.1	38	17.8	10.64
11	Total head (“a” in Figure 1) m					
	max	16.08	16.72	18.34	18.4	14.7
	min	15.91	16.47	16.87	16.9	13.2
12	Linear rod power at 5% overall power$^{(*)}$ (K_v/m)	$1.27^{(\circ)}$	1.08	0.86	0.91	1.12
13	Actual $K_v^{(+)}$	1/640	1/619	1/144	1/48	1/2.5

$^{(*)}$The % value is the power related to the reported K_v.
$^{(+)}$Related to LSTF.
$^{(\circ)}$To compensate heat losses.

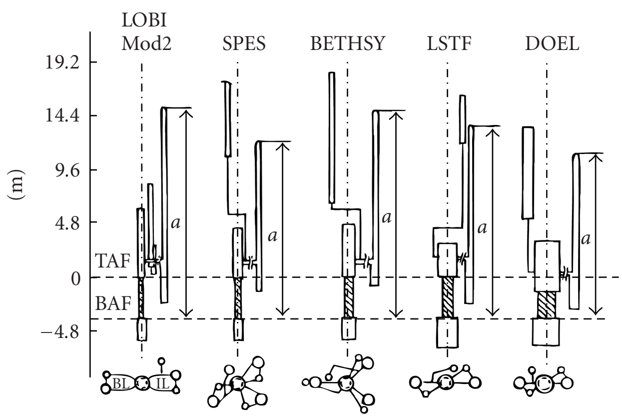

Figure 1: Sketch of the facilities considered for the experimental data base evaluation.

its search of understanding the physical models pertinence. A first group of indications is given by the analysis of the discrepancies between calculations and experimental results, which has been discussed above. A second group relates to the discrepancies between the results of different codes. This last group is often very valuable because the differences of models between the codes can be quite easily identified. Consequently, the analyst can focus immediately on the concerned physical models and evaluate their relative capabilities in reference with the experimental data. By the wide variety of codes used, ISPs give good opportunities for doing extensive analysis of this kind.

(iii) The last category of comparisons, which ISPs allow, is the comparison of the results obtained with the same code by different users. The major differences between the calculations with the same code can be mainly attributed to the users of the code and this effect has been called the “user effect.” Indeed this effect is a major finding of ISPs activity. It has been discovered very early by running the very first ISPs on thermal-hydraulics. The development of thermal-hydraulic advanced codes was expected to decrease this effect, but the last thermal-hydraulic ISPs have shown that there was still a significant “user effect” with these advanced codes. Detailed studies of this effect have been made on different ISPs and especially on ISP 26 [11]. In addition to the identification of the user effect, ISPs have contributed largely to its understanding. ISPs are really providing data, which are absolutely unique on this crucial subject. Even though some suggested ways to reduce the user effect have been proposed, it remains that we are quite far from controlling it. This user effect has also appeared as a generic question and not only in the thermal-hydraulics area where it has been discovered. In particular the several ISPs, which have been recently performed in the severe accidents area, have shown the importance of such an effect.

In the coming sections, specific analysis and further discussions will be provided on selected small break LOCA and transient ISPs.

6. OUTLINE OF INVOLVED FACILITIES AND TESTS FOR SB-LOCA ISPs

6.1. Facilities and plant hardware

In this section, information is given concerning some hardware features that are relevant for the considered ISP tests. Figure 1 shows the sketch of LOBI, SPES, BETHSY, and LSTF facilities and of the Doel plant.

The relative elevations of important system components like core, steam generators U-tubes, loop seals can be seen; the number of loops constituting the system is reported too. The most important design parameters of the considered facilities and of the plant are given in Table 2. All the considered facilities can operate at the reference plant nominal pressure for both primary and secondary loops. The height scaling ratio is equal to one in all cases, so the gravity heads are properly simulated. The maximum allowed power is equal to the reference reactor value multiplied by volume scaling ratio only in the cases of LOBI and SPES. In other cases, a decay power value is allowed, ranging around 10% of the nominal value. This scaling limitation prevents, among the other things, the possibility to have simultaneously rightly scaled temperatures and flowrates in nominal conditions. In these facilities, the choice is generally made to preserve hot leg fluid temperature during steady state operation, before any transient; alternatively, it is possible to preserve the cold leg fluid temperature and nominal flowrate (hot leg temperature not preserved); as a consequence of the former choice, secondary side fluid temperature and pressures must be higher than the reference plant nominal values (a real plant at hot standby conditions, 10% of nominal power, exhibits the same behaviour, roughly 70 bar at secondary side); still, primary pumps have not the maximum allowable flowrate and head properly scaled, although in the case of BETHSY, primary pumps have full flowrate capacity and preserve the head in single phase flow conditions. The different criteria utilized for the pressurizer result from Figure 1, as well for defining the minimum elevation of the loop seal. In the facilities (SPES, BETHSY, and LSTF), the $L/\sqrt{D}$ scaling is adopted for the design of hot and cold legs piping also preserving the volume scaling [14].

Nevertheless, the position of the hot leg axis with respect to the top of the active fuel may be not the same as in the reference nuclear power plant; in BETHSY, this position is preserved with respect to the reference reactor, as well the bottom line of the cold leg elevation to the bottom of active fuel, this leads to different elevations for hot and cold leg axes. For all the multiloop facilities, each primary (and secondary) circuit is equal to the other; thus nearly symmetrical thermal-hydraulic conditions occur in the various loops. An exception is represented by LOBI, where one loop (intact) simulates three loops of the reference reactor and the other simulates a single (broken) loop. Hardware parameters like pump geometrical configuration, presence, and characteristics of bypass flow paths (mostly in the vessel) can play an important role in the considered test scenarios.

6.2. Outline of the experimental scenarios

The experiments A2-81, SP-FW-02, SB-CL-18, 9.1b, and the SGTR transient, respectively from LOBI, SPES, LSTF, and BETHSY facilities and Doel plant (Figure 1 and Table 2), were submitted by the facility owner organisations to the CSNI and were discussed and approved at working group and principal working group levels. The list of host organisations (i.e., proposing the exercise, writing the final reports, and chairing the workshops) for each ISP, is given in Table 3.

Table 3: List of host organisations for small break LOCA related ISPs.

ISP	Host organisation
18	JRC (Ispra)
20	TRACTEBEL (Brussels)
22	ENEA (Rome)
26	JAERI (Tokai Mura)
27	CEA/CENG (Grenoble)

The procedures outlined in [3] for assignments of ISPs have been generally followed.

The main characteristics of the mentioned tests are reported in Table 4. The main phenomena occurring during SB-LOCAs are listed in Table 5 [15], making use of a phenomena matrix developed in state-of-the-art report (SOAR) on emergency core cooling thermal-hydraulics [15]. In the same table, a qualitative evaluation of the capabilities of facilities is provided, according to three judgment levels. For completeness and in order to give an example of the possible use of this table, in the last two columns, an overall evaluation of the Relap5/Mod2 and CATHARE codes in addition to their performances is reported, considering each of the phenomena listed and the pre- and posttest calculations [15].

The significant trends of variables with reference to the selected tests are shown in Figures 2 through 7, while details of the experiments are given below.

ISP 18: The test in LOBI simulated a 1% cold leg break with HPIS intervention (Figure 2). From a phenomenological point of view, the whole transient can be divided into three main phases:

(i) the forced circulation period,
(ii) the two-phase natural circulation period,
(iii) the reflux condensation period.

During the first phase, after the opening of the break device, the primary system pressure decreases down to 13.2 MPa within 32 seconds, triggering both SG isolation and core power decay. Simultaneously, secondary system cooling is activated causing an upper limit to the increase in secondary pressure. At 45 seconds pumps coast down begins and at 74 seconds HPIS starts to inject water into the primary system. At 121 seconds pump coast down completion ends the forced circulation phase, and two-phase natural circulation is established in the loops. As voiding proceeds, natural circulation stops and heat exchange with the secondary system is accomplished by reflux condensation occurring in the steam generator U-tubes.

An important feature of the test is the liquid mass distribution inside the primary loop which is affected by the bypass flow paths in the vessel and by heat transfer across steam generators mainly during natural circulation and reflux condensation periods. Since HPIS is sufficient to avoid core uncovery, no dry out is measured during the test.

ISP 20: The considered transient in Doel plant is the steam generator tube rupture (SGTR) accident (with a longitudinal crack of 7 cm long located in the ascending leg of

Table 4: Main characteristics of the considered transient.

ISP ident.	Facility/plant	Test	Type	Secondary side significant conditions	Emergency systems in primary side(*)	Recovery procedure	End of test (s)
ISP 18	LOBI	A2-81	SBLOCA A_b = 1% A_{max} in cold leg	Imposed 100K/hr	HPIS in cold leg	Secondary system feed and bleed	4500
ISP 20	DOEL	SGTR 1979	SGTR $A_b = 0.5^{(+)}$% of A_{max}	EFW and steam Relief valves active	HPIS in cold leg: pressurizer sprays and heaters	—	3000
ISP 22	SPES	SP-FW-02	LOFW-loss of feed water	Boildown of secondary side and EFW active in one loop	Pressurizer PORV and heaters	EFW in one loop	8000
ISP 26	LSTF	SB-CL-18	SBLOCA A_b = 5% of A_{max}	Steam relief valves active	Accumulators and LPIS in cold leg	RHR actuation	1000
ISP 27	BETHSY	9.1.b	SBLOCA A_b = 0.4% of A_{max}	EFW and pressure control active	Accumulators and LPIS in cold leg	Depressurizarion of secondary side	8000

(*)scram is assumed in all cases following a low pressure signal.
(+)rough evaluation

the U-bend of one of the U-tubes) occurred in Doel plant in 1979 and constituted the first (and, so far, the unique) standard problem related to a plant system (Figure 3). At the moment when the event occurred, the reactor was subcritical with all control rods down and the pressurizer heaters on. In the secondary side, the steam lines were both isolated by the MSIV and no condenser vacuum was available. The main feed water pumps were not operational and water level in both SGs was manually controlled by means of a letdown system. The auxiliary feed water pumps were not running. The plant conditions remained well below the safety margins during the whole transient.

The condensation induced by the pressurizer spray and in the secondary side of steam generators at the liquid-steam interface is the relevant phenomena to be predicted by codes. However, quite large uncertainties characterize the trends of the main quantities as well as the time of actuation of the main systems, typically reflecting the features and capabilities of plant instrumentation and recording systems.

ISP 22: The test in the SPES facility consists of a loss of feed water with delayed actuation of emergency feed water in one of the three loops of the facility. The transient evolves through 5 phases (Figures 4 and 5) from the following.

(i) The accident beginning to scram: due to the loss of feed water, the downcomer level drops quickly in each steam generator. As the low level set point is achieved, the scram occurs, causing the core power to shutoff and the main steam isolation valves to close.

(ii) Scram to pressurizer PORV opening: after scram a quick depressurization occurs in primary side as a consequence of temperature decrease. The steam generators U-tubes then dry out, the primary temperature rises continuously, causing primary system pressurization up to the pressurizer PORV opening.

(iii) Pressurizer PORV opening to pumps trip: while the primary temperature is rising continuously and is approaching the saturation value, the pumps are switched off when the fluid subcooling at the inlet reaches the set point value.

(iv) Pumps trip to emergency feed water activation: due to the progressive voiding of the primary side, a core heat up occurs and the emergency feed water activation signal in one of the steam generators is generated by the high rod surface temperature set point.

(v) Emergency feed water activation to the end of the transient: emergency feed water activation causes a quick repressurization in the affected steam generator and reestablishes heat transfer between the primary and the secondary sides, with a consequent big decrease of primary temperature and pressure. The secondary level in the affected steam generator increases steadily until the initial value is restored.

The following main features of the test can be pointed out.

(i) The pressure control of the primary system by the pressurizer PORV cycling and the consequent mass depletion cause rod surface temperature excursion roughly two hours after the transient beginning.

(ii) The actuation of emergency feed water in one loop leads to primary system depressurization, pressurizer draining, core quench, and brings the facility to safe shut down conditions, allowing the possibility of accumulators actuation.

ISP 26: The experiment in the LSTF test facility is originated by a 5% break in the cold leg of the loop without pressurizer, the HPIS is not available (Figure 6). Following the break opening the primary pressure went down and scram occurred at 9 seconds. The core was temporarily uncovered, at first time, between about 120 and 155 seconds after break opening. The reason for this was a core level depression amplified by a manometric effect caused by condensation at the top of U-tubes and consequent liquid holdup in

TABLE 5: Suitability of tests facilities, judgment of the experiments, and (example of) evaluation of RELAP5/Mod2 and CATHARE code capabilities as from [13].

	Facilities				Experiments						
Phenomena	SPES	LOBI/Mod2	BETHSY	LSTF	ISP 18	ISP 20	ISP 22	ISP 26	ISP 27	RELAP5/Mod2 Code Performance$^{(\circ)}$	CATHARE V 1.3 Code Performance
Natural circulation in one-phase flow, primary side	+	+	+	+	o	–	–	–	–	+	+
Natural circulation in two-phase flow, primary side	+	+	+	+	+	–	o	–	o	+	+
Reflux condenser mode and CCFL	o	o	+	+	o	–	+	+	+	o	o
Asymmetric loop behaviour	+	o	+	o	–	o	+	o	+	o	o
Leak flow	o	+	+	+	–	–	o	+	+	o	+
Phase separation without mixture level formation	+	+	+	+	–	–	–	–	o	×	+
Mixture level and entrainment in steam generator secondary side	o	o	o	+	o	o	+	o	o	o	o
Mixture level and entrainment in the core	o	o	+	–	–	–	+	+	+	o	o
Stratification in horizontal pipes	o	o	+	+	o	–	–	+	+	o	+
Emergency core cooling mixing and condensation	o	o	o	+	o	–	–	+	+	–	o
Loop seal clearance	+	+	+	+	–	–	–	+	+	+	+
Pool formation in upper plenum/CCFL	o	–	o	o	–	–	–	o	o	–	o
Core wide void and flow distribution	–	–	–	o	–	–	–	+	o	–	×
Heat transfer in covered core	+	+	+	+	o	–	+	+	+	+	+
Heat transfer in partially uncovered core	o	o	+	+	–	–	+	+	+	o	o
Heat transfer in steam generator primary side	+	+	+	+	+	–	+	+	+	+	+
Heat transfer in steam generator secondary side	o	+	+	+	o	–	+	o	+	+	+
Pressurizer thermal-hydraulics	o	o	o	o	–	o	+	o	o	+	+
Surge line hydraulics (CCFL, chocking)	o	o	o	o	–	–	+		–	o	o
One and two-phase pump behaviour	o	+	+	+	–	–	o	–	o	+	+
Structural heat and heat losses	o	o	o	o	o	–	–	o	o	+	+

Table 5: Continued.

Phenomena	Facilities				Experiments					RELAP5/Mod2 Code Performance(°)	CATHARE V 1.3 Code Performance
	SPES	LOBI/Mod2	BETHSY	LSTF	ISP 18	ISP 20	ISP 22	ISP 26	ISP 27		
Noncondensable gas effect on leak flow	−	+	o	o	−	−	−	−	−	×	×
Phase separation in T-junctions	o	o	+	+	o	−	−	+	+	−	+
Thermal-hydraulic nuclear feedback	−	−	−	−	−	−	−	−	−	×	×
Boron mixing and transport	−	−	−	−	−		−	−	−	×	×
Separator behaviour	−	−	−	−	−	−	−	−	−	×	×

(°)The best performance of the code is considered due to number of submissions.
Note: the following symbols are used in this table; for test facility versus phenomenon: + suitable for code assessment, o limited suitability, − not suitable, for phenomenon versus experiments: + experimentally well defined, o occurring but not well characterised, − not occurring or not measured, for phenomenon versus code calculation: + well predicted, o qualitatively predicted, − not predicted, × not applicable.

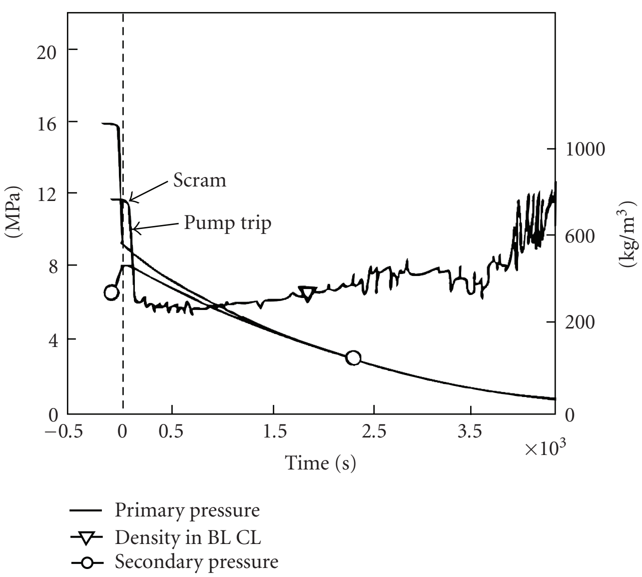

Figure 2: ISP-18 (LOBI): experimental trends of primary and secondary side pressures and broken loop cold leg density.

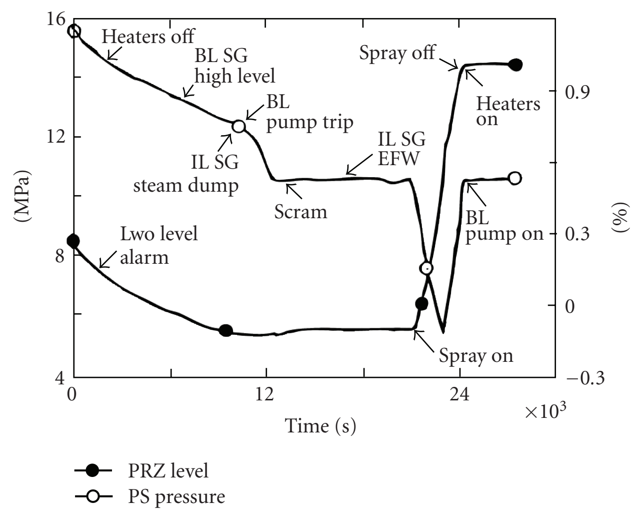

Figure 3: ISP-20 (Doel-2): registered data trends of primary side pressure and pressurizer level.

the ascending and descending legs of U-tubes. At about 140 seconds, loop seal clearing occurred and caused a temporary core temperature recovery. After loop seal clearing, the break flow changed from low quality to high quality two-phase flow and the depressurization of primary loop was accelerated. By about 180 seconds after the break, the primary loop pressure decreased below steam generator secondary side pressure. Thereafter, the steam generator no longer served as heat sink and the energy removal from the primary system occurred through the discharge of coolant from the break. It is noted that loop seal clearing occurred before the reversal in primary and secondary pressures. The core was uncovered again after about 420 seconds due to vessel inventory boiloff; the heater rods in the upper part of the core showed superheating up to about 80 K. The core was covered with two-phase mixture again after about 540 seconds by the accumulator water injection. The peak cladding temperature in the test was approximately 740 K, observed during the temporary core uncovery just before the loop seal clearing.

The occurrence of two dry out and quench conditions constitutes the main peculiarity of this transient. The mass distribution in the loop and the heat transfer with secondary side constitute further challenging phenomena for code assessment.

ISP 27: The test in BETHSY facility is an SBLOCA with the break (roughly 0.5%) located in the cold leg of the loop

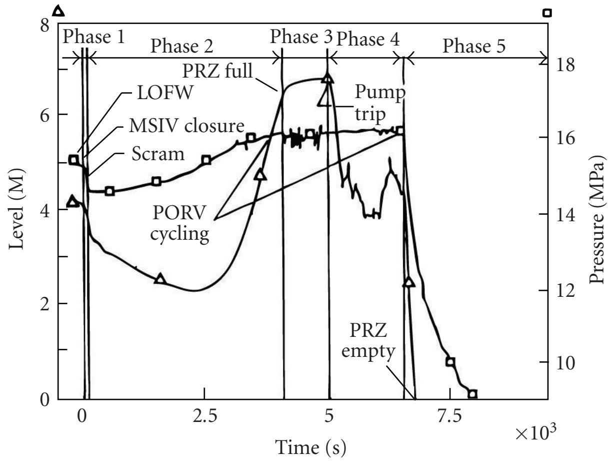

FIGURE 4: ISP-22 (SPES): experimental trends of pressurizer pressure and level.

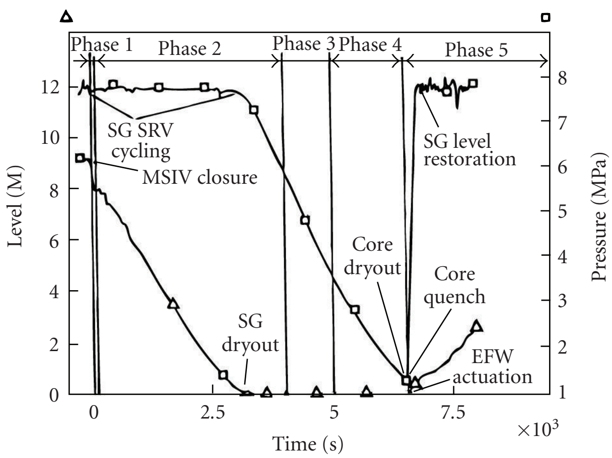

FIGURE 5: ISP-22 (SPES): experimental trends of steam generator pressure and level.

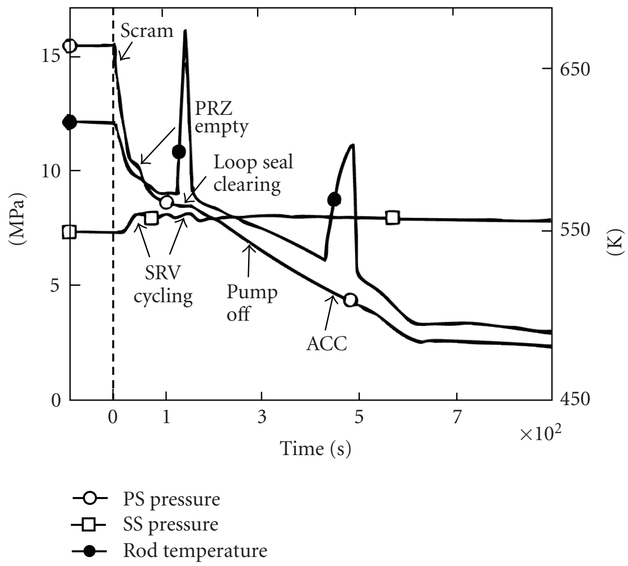

FIGURE 6: ISP-26 (ROSA-IV): experimental trends of primary and secondary side pressures and rod surface temperature.

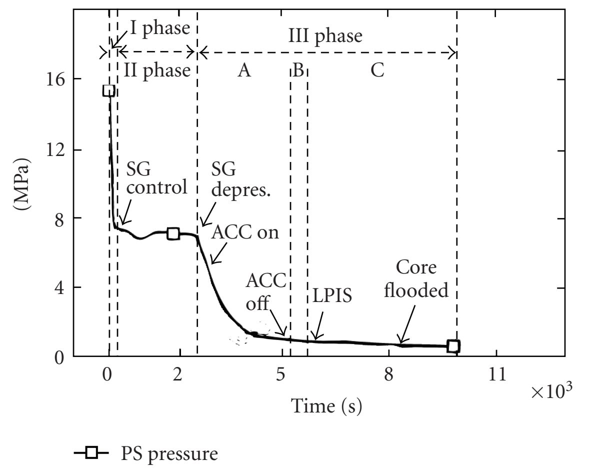

FIGURE 7: ISP 27 (BETHSY): experimental trend of primary side pressure.

with the pressurizer (Figure 7); HPIS is not available. Three different phases can be recognized during the transient:

(i) subcooled blowdown;
(ii) mass depletion in primary side;
(iii) ultimate procedure.

Subcooled blowdown

Following the break opening the primary pressure falls down and scram occurs when the pressure reaches 13.1 MPa. safety injection signal (SI) occurs at 11.9 MPa. Following SI signal, turbine bypass occurs and main feed water is off. Before SI, secondary side pressure is controlled through the spray condenser and remains constant at 6.91 MPa; when turbine bypass occurs the pressure threshold becomes 7.03 MPa. Auxiliary feed water injection starts 30 seconds after SI signal, and pump coast down initiates 300 seconds after the same signal. During this phase, the pressurizer and surge line empty leading to the relatively fast depressurization of the primary side; in the same period owing to the diminution of the heat transfer from primary to secondary side, the mass flowrate in the secondary side starts to decrease.

Mass depletion

The second phase is characterized by mass depletion and almost constant pressure and temperature in primary loop (saturation values). Oscillations in break flowrate in the first period of phase 2 testify of little voiding of the cold leg of the broken loop. Later on, with pumps at rest, once the upper head to downcomer bypass steam flows to the broken cold leg, mostly steam flows at the break (stratified conditions with liquid level upstream the break lower than the elevation of the exit nozzle axis). Loop seal clearing is recognized to appear in only one of the two intact loops and stops with the occurrence of the first core uncovery. Secondary side conditions (mostly levels) remain constants in this period. At the end of this phase, a second core uncovery occurs, which causes the trip for the predefined ultimate procedure when the core maximum clad temperature reaches 723 K.

The ultimate procedure

This phase of the test consists in fully opening the dump valves in secondary side due to accumulators and LPIS actuation; three different parts can be distinguished during the last phase of the transient (A, B, and C, resp.). In the part A, starting with the ultimate procedure initiation and ending with accumulators isolation, intense condensation in the U-tubes induces liquid fall back to the core, which is cooled from the top, then accumulator injection allows the clad temperature to turn around and the core to be rewetted. Part B is related to the period from the accumulators' isolation up to LPIS actuation. A continuous mass depletion of primary side without ECC injection characterizes this phase. No dry out situation occurs in this period during which the primary pressure decreases down to achieving the set point for LPIS actuation. Very early during part C, LPIS flowrate becomes larger than break flowrate leading to recover the primary coolant system. In this period filling up the primary loop occurs causing, among other things, direct contact condensation between the cold liquid injected by LPIS and the steam present in the primary loop.

7. THE RESULTS OF SOME STATISTICAL ANALYSIS FOR SMALL BREAK LOCA ISPs

In the framework of the ISP activity evaluation, interesting information may come from the statistical analysis considering the number of participants to the ISP, including countries and organisations, as well as the adopted thermal-hydraulic system codes. The main goals of the effort are to get an overview of the interest towards the ISP activity from the international scientific community, and to derive information about the engagement by different organisations in the use of large thermal-hydraulic system codes.

A wide database is available for making statistical evaluations; this is included in the ISP reports approved and issued by CSNI and in the individual ISP participants written contributions normally distributed (among participants) at the ISP workshops. A comprehensive analysis would require establishing homogeneous indices for interpreting the data, for example,

(i) computers have strongly evolved lowering the needed calculation time in the period 1985–1995 (in some cases, the calculation time increases just because transients take longer times);
(ii) codes having sophisticated capabilities of noding a specific zone of nuclear power plant (i.e., volume component in CATHARE) may need less overall number of node for having the same detail of plant description;
(iii) once an acceptable convergence is reached from a numeric point of view, the increase in number of time steps might not lead to any benefit; calculation time may be reduced by the progress in physical modelling reducing the interaction number and meshing size.

However, a number of quantities could be used to characterize the results of an extended statistical analysis, for example, [16]. Following a discussion among the participating working group members, it was found that most of the data (e.g., numbers of used meshes or nodes) averaged on the number of participants could be misinterpreted or even misleading considering the present situation. This is originated by the reason outlined above, specifically, including the different levels of qualification of the scientists directly involved in the calculation and even the different purposes for organisations in participating in an ISP. As an example, it was found that the consideration of the number of input deck nodes for the different participants should not give a reasonable index of the "quality" of user nodalization itself.

The lack, in the ISP documents of an exhaustive description of calculation resources, prevented the possibility to use the time needed for the calculation of ISP exercises, as a parameter eventually identifying a "code speed" index.

Keeping in mind the above, the following quantities were selected for the present analysis:

(i) kind and number of participants to the ISP,
(ii) thermal-hydraulic codes used for the ISP calculation.

In relation to the first item, it seemed interesting to correlate the participants with the different ISPs and with the adopted codes used, considering the total number of participations to the ISPs for each participant.

The second item gives an idea of the differences in the use of each code. It must be emphasized that the results of the analysis might not be indicative of the actual number of users for each code. More detailed information in this context should be gathered by specific collaborative programmes like Club des Utilisateurs du CATHARE (CUC), Code Assessment and Maintenance Program (CAMP) or specific "institutionalized" series of conferences like Relap5 International Conference.

Specific parameters to characterize the two items identified above, which were retained suitable for evaluating the overall impact of ISP activity in the scientific community are

(1) number of participants to the specific ISP,
(2) participants per ISP,
(3) number of countries per ISP,
(4) participants per code per ISP,
(5) codes used per ISP.

ISP phases (e.g., pre- and posttest) are considered in Tables 6, 7, 8 and the information related to items (1) to (5) are given in these tables. As already mentioned, further information on statistical evaluation, considering a large number of parameters, can be found in [4, 16]. It is to be noted that there are six types of organisations who participated in the small break LOCA ISP exercises. These are covering a wide range of organisations: research centres, universities, licensing

Table 6: Participants per code per ISP.

	ISP 18 pretest	ISP 18 posttest	ISP 20 posttest	ISP 22 pretest	ISP 22 posttest	ISP 26 posttest	ISP 27 pretest	ISP 27 posttest
RELAP4/Mod6	2	1	—	3	2	—	1	1
RELAP5/Modl	10	1	1	1	—	—	—	—
RELAP5/Mod2	4	4	5	4	1	7	10	11
RELAP5/Mod2.5	—	—	—	—	—	2	1	1
RELAP5/Mod3	—	—	—	—	—	—	7	6
CATHARE 1	1	1	—	—	—	—	—	—
CATHARE 2	—	—	1	1	2	1	3	6
TRAC/PF1	3	1	—	4	—	1	1	—
DRUFAN-M2	3	—	—	—	—	—	—	—
ATHLET 1.0	—	—	—	—	—	1	1	1
SMABRE	—	—	1	1	—	3	—	—
NOTRUMP	1	—	—	1	—	2	1	1
FRACAS	1	—	—	—	—	1	—	—
Dinamyka	—	—	—	—	—	—	1	—
Tech-m4	—	—	—	—	—	1	1	—
Moot	—	—	—	1	—	—	—	—
ATHENA	1	—	—	—	—	—	—	—
SATAN-M	—	—	—	—	—	1	—	—

authorities, industry, utility, and others (e.g., engineering companies).

Detailed statistical data and analysis are included in [4]; in this paper, a few conclusions drawn from the analysis of the statistical data are given as follows.

(i) A large number of codes have been used in the different ISPs. It is possible to see a predominant use of RELAP family of codes specifically from most of universities and research centres.

(ii) A number of participants still use first generation (e.g., RELAP4) or proprietary codes (NOTRUMP).

(iii) The number of participants increased after ISP 20 essentially due to the fact that since the time of ISP 22, the ISP activity was open for the non-OECD countries. The positive effect was to allow Eastern countries to get information about Western countries safety methodologies. A "negative" impact of this was the increment of the scientists participating to the ISP for the first time, making more difficult to get objective conclusions from the discussions about the ISP itself.

(iv) The use of well established or "frozen" versions of codes allows the verification of the degree of assessment of the concerned code version against a full transient.

(v) Fourty six organisations took part in the small break LOCA ISP activities; very few organisations took part to more than five of the considered ISP cases.

(vi) Of the above organisations, almost 82% belong to the research/university side (specifically, 54% research institutes and 28% universities).

8. SOME LESSONS LEARNED FROM THE SMALL BREAK LOCA ISP ACTIVITY

The contents of this section are based on the answers received to a questionnaire [4] that was sent to fourteen members of TG-THSB who were involved in the analysis of most of the small break LOCA ISPs, from the conclusions included in each of the ISP final report (CSNI reports, [5] to [10]), and from the discussions of a working group, which took place during the meeting in Pisa University in 1995.

As mentioned in Section 7, eighteen different codes were used by the participants for these ISPs. It is not the purpose here to produce a detailed analysis of calculational performances, code by code, and ISP by ISP; but in a more synthetic approach, to derive the main outcomes from the five ISPs, specifically taking into account the following four items identified in the questionnaire:

(i) code deficiencies and capabilities,
(ii) progress in the code capabilities,
(iii) possibility of scaling,
(iv) other comments.

It should be mentioned that from ISP 18 to ISP 27, more and more physical phenomena were involved in the transients which were dealt within the ISP exercises, such as core-uncovery and heatup, pressurizer discharge, secondary side voiding and filling, low pressure two phase flows as well as interacting operator actions. The involvement of various phenomena during an ISP exercise must be considered as challenging for the codes, and as well as code users. Furthermore, increasing overall complexity and longer time durations of the transients to be calculated, can be noted during

Table 7: Countries, Participants, and Codes used per ISP.

ISP	Type	No. of participants	No. of countries	No. of codes
ISP 18	Pretest	27	11	8
ISP 18	Posttest	6	6	5
ISP 20	Posttest	7	5	4
ISP 22	Pretest	17	14	8
ISP 22	Posttest	4	4	3
ISP 26	Posttest	17	14	9
ISP 27	Pretest	23	17	9
ISP 27	Posttest	17	14	7

(∗) Note: the numbers reported in the number of participants column do not coincide with the sum of the numbers in Table 6 because either a single participant might have submitted more than one calculation officially recognized, or a group of participants took part together in the ISP submitting a single calculation.

Table 8: Calculations per code groups per ISP.

Codes	ISP I8 pre	ISP 18 post	ISP 20 post	ISP 22 pre	ISP 22 post	ISP 26 post	ISP 27 pre	ISP 27 post
RELAP family of codes	16	6	6	8	3	9	19	19
CATHARE family of codes	1	1	1	1	2	1	3	6
TRAC family of codes	3	1	—	4	—	1	1	—
Others	6	—	1	3	—	9	3	2
Total	26	8	8	16	5	20	27	27

the process of going from the earlier to the latest considered small break LOCA ISPs.

8.1. Code deficiencies and capabilities

The code user is clearly the best judge of the performance of his own calculations. The invested resources, the depth of the quality assurance used when setting up the nodalization, and the possibility to interact with the experimentalists play a major role in the quality of the results, this can only be known to the user. So, in order to get a general, but not in depth evaluation of submitted results, two steps were considered as follows:

(a) list of relevant thermal-hydraulic phenomena in each test, making reference to the list in Table 5, also looking at the facilities suitability;
(b) identification of phenomena which were not well predicted by the majority of submitted calculations.

The quality of experimental data also had a role in selecting code deficiencies. A list of generic code deficiencies, which were identified, is provided in Table 9. As code deficiency, it was meant a situation where either the phenomenon is not predicted to occur in the calculation, or the phenomenon was predicted but at a given time the quantity $|Yc - Y_E|/|Y_E|$ was larger than 0.20 (see also [9]). In this case, Y is a relevant thermal-hydraulic quantity representing the assigned phenomenon and the deviation of calculated from experimental quantity.

It can be seen from Table 9 that thirteen main code deficiencies have been found, some of those being common to different ISPs. A comprehensive and systematic qualitative or quantitative code calculation accuracy evaluation is well beyond the scope of the present paper. In this respect, some example results are provided in [8, 9, 17] in relation to ISPs22, 26, and 27, respectively. Slightly different criteria are adopted for achieving either a qualitative judgment (e.g., good, average, and poor) or a quantitative evaluation (e.g., quantification of the accuracy through the fast fourier transform- (FFT-) based method). For this type of evaluations, the interested researcher could refer directly to the mentioned documents. Additional notes on selected items are provided below.

Let us first deal with the break flowrate problem (item 1) in Table 9 appearing in all ISPs, but not in ISP 20 and 22; many participants have experienced wrong predictions of this parameter among the ISPs, leading to deviation (sometimes large) from the actual transient. Although a very accurate prediction of this quantity is not requested for safety studies, where a stated range of break flowrate may be and is generally used, the capability of codes to reasonably predict two-phase critical flowrates versus leak geometry and upstream conditions becomes significant when the efficiency of operator actions (use of discharge devices, e.g.) has to be investigated. For the considered ISPs, various levels of agreement on the break flowrate predictions were observed, and these results were often correlated with the resources invested in this part of the work and the user's experience in this field. It appears however that some break models are still having difficulty to calculate for the whole range of break upstream conditions. In this area, an example of complex interaction between code nodal inadequacies, user assumptions,

Table 9: General code deficiencies for the considered ISPs.

ISP no.	Identification no. of Deficiency	Code deficiencies
18	(1)	Break flow
	(2)	Stratification in cold and hot leg
	(3)	Mass distribution in primary side
	(4)	Mixing in the downcomer
20	(5)	Steam condensation
	(6)	Level simulation in secondary side
22	(6)	Unability to predict mass inventory in secondary side
	(7)	Heat transfer between EFW and hot SG walls
	(8)	Pressurizer behaviour including PORV leak
	(9)	Coolant mass distribution in primary circuit
	(1)	Break flow
	(2)	Stratification in cold and hot leg
26	(10)	CCFL in SG plena
	(3–11)	Core level depression
	(12)	Core uncovery and heat-up
27	(1)	Break flow
	(2)	Stratification in cold and hot leg
	(3)	Mass distribution in primary side
	(4)	Mixing in the downcomer
	(12)	Core uncovery and heat-up
	(13)	Low pressure period

interpretation of data provided by experimentalists is given in [18] by using the RELAP5 code. This sensitivity study about break discharge coefficients, performed during the ISP 27 posttest analysis, showed the large influence of this parameter upon the time scale shifting appearing in blind calculations. Even though, these coefficients had been previously adjusted by using the separate effect test experimental data provided by the ISP host organisation. This mentioned study pointed out and also emphasized the need for code assessment procedures to verify the overall agreement on integral test transients.

However, in general, break flow can be largely influenced by the upstream flow conditions, which are strongly related to the mass distribution in the entire system and to the overall system behaviour. Therefore, just “tuning” the break flowrate might introduce a compensation of errors and, as well as, it might result in complete wrong conclusions. This also results in excluding to provide the ISP participants with the measured break flow. For complicated geometries (such as valves), geometry effects on break flow are even more important. The critical flow performance of the valves must be characterized and supplied as input to the code.

Another key parameter in these considered ISPs is the coolant mass distribution in the primary circuit (item 3 in Table 9, relevant to ISPs 18, 22, and 27), which is strongly related to the two-phase structure and flow regimes. Interfacial shear stresses, counter-current flow limitations, transitions between flow regimes are directly related to the coolant mass distribution. The need for a better prediction of this distribution prompted the development of second-generation (“advanced”) two-phase thermal-hydraulic codes. These codes proved their ability to qualitatively predict the physical phenomena involved during the different transients, such as stratified flows in horizontal pipes, loop seal clearing, interfacial transport in core, and steam generator U-tubes. Nevertheless, some weaknesses revealed during the first of the considered ISPs and, concerning void distribution in vertical or horizontal components, still appeared unresolved in ISP 27 (see Table 9).

Additional specific comments are connected with the thermal coupling between fluid and structures, both in primary and secondary sides. This is a consequence of both the scaling ratio of the facilities involved, and of the operating procedures applied; this has been a subject of discussion during most of the ISP related workshops. Inaccuracies due to different reasons in accounting for the fluid structure and thermal coupling, that is, lack of suitable noding and inadequate consideration of heat losses, may have a role in various calculation discrepancies. In every case, codes have demonstrated their ability to qualitatively describe these phenomena (fluid-structures heat transfer), provided that a sufficient amount of care and work had been spent to correctly define the geometry and thermal boundary conditions.

In ISPs 26 and 27 discrepancies remain in predicting core heatup, though fluid distribution is predicted adequately. Similarly “hot wall delay” effect in steam generators downcomer is not satisfactorily calculated in ISP 22. These examples raised questions about the relevant heat transfer models in the considered conditions.

At last, some specific aspects specifics for one or two ISPs, such as secondary side level prediction (ISPs 20 and 22), and low pressure refilling of the primary coolant system (ISP 27), highlighted model weaknesses in these fields for most of the codes.

From the point of view of the code capabilities, it must be indicated that experienced users are able to get the relevant phenomena even in the case when complex scenarios are involved. Such a qualitative judgment has been supported by quantitative evaluations, that is, quantification of accuracy considering experimental and calculated trends, in the cases of ISP 22 and ISP 27 (see also below).

However, looking generally to a single ISP, a wide range of results is achieved even considering the use of same code versions. This emphasizes the role of the user in setting up the nodalization and also in interpreting the initial and boundary conditions supplied by the experimentalists. In conclusion, in an ISP framework, owing to different reasons (see also below) the user effect may overshadow the reasons for code deficiencies, thus preventing the possibility to identify code capabilities

8.2. Identification of progress in code capabilities

Firstly, it must be emphasized that one of the reasons why progress is difficult to measure, is that it is difficult to isolate phenomena in an integral test. Owing to this fact, it is also difficult to judge even making reference to each single code, since there is also no clear feedback between the ISP activity and the code developers, as already mentioned. In fact, ISPs have been proved more useful to provide information on the capabilities of the thermal-hydraulic codes, especially when posttest calculations or parametric studies were conducted, than to identify the deficiencies or failures. In this case, returning to the use of more analytical work or separate effect tests is however necessary to modify or extend the individual physical models; this step has allowed some progress in code capabilities. The direct contact condensation, or stratification and phase separation models in horizontal pipes constitute an example of this.

Progress was also observed in using parallel channel simulation in attempting to better represent 2D or 3D behaviours with the codes used, which are basically one dimensional. One of the most important progresses has been obtained in the area of users guidelines. Thanks to the large number of participants, often using the same code versions, with different nodalizations and option choices, the ISP pre- and posttest calculations, formed a wide "database" for the so called "user effect."

The small break LOCA ISPs provided a useful information basis, not only for experienced code users to increase their capability from one ISP to the other, but also for new code users to improve their know-how by exchanging ideas and meeting more experienced people in the frame of ISPs.

8.3. Possibility of scaling

Although the considered five ISPs address the problem of scaling, either because the plant transient is expected to be very similar to that observed in the facilities which are properly scaled, or because of the different scales of the facilities addressing the same thermal-hydraulic phenomenon, or because a plant transient is considered (ISP 20), the commonly reached conclusion is that small break ISPs alone are not sufficient to check code accuracy in this field. The counterpart tests performed making reference to the same scenario in terms of boundary and initial conditions, on different facilities, are much more valuable for this task [17, 19, 20].

However, it is considered interesting to bring to the attention hereafter the results of a common evaluation, which was made in preparing CSNI report on "lessons learned from OECD/CSNI ISP on small break LOCA" [4].

Two items are identified to judge the possibility of using the small break LOCA ISP exercises in scaling activities.

(A) Realism of involved physical phenomena as far as plant is concerned.
(B) Possibility to assess the code in different scaled facilities in relation to the same scenario (evaluation whether the small break LOCA ISP scenario can be found in different scaled facilities).

The analysis of each small break LOCA ISP related to the above two items gives the following results.

(i) *ISP 18*, item (A): test scenario expected to be similar in the plant.
(ii) *ISP 18*, item (B): limited suitability because the test scenario not available in other facilities.
(iii) *ISP 20*, item (A): this is a plant scenario.
(iv) *ISP 20*, item (B): the same scenario has been considered in one of the LOBI experiment.
(v) *ISP 22*, item (A): qualitatively, phenomena expected to be the same as in the plant, but timing is different.
(vi) *ISP 22*, item (B): test suitable for scaling because the same experiment was repeated in different facilities.
(vii) *ISP 26*, item (A): plant scenario expected to be the same (local phenomena might be different).
(viii) *ISP 26*, item (B): test suitable for scaling because the counterpart test activity deals with similar scenario.
(ix) *ISP 27*, item (A): plant overall scenario expected to be the same.
(x) *ISP 27*, item (B): difficult to assess the code scaling capabilities, because the similar test scenario is not available from other facilities.

As a result of the above, ISP 22 and ISP 26 related experiments appears to be the most suitable for studying scaling. Even though it is a plant, ISP 20 mostly suffers of limitations due to inadequacy of the database obtained from the plant, both in relation to plant hardware and data recording, as already mentioned.

8.4. Other comments

An additional outcome from the small break LOCA ISP activity in the second half of 90s appeared is linked to the area of works about quantitative accuracy evaluation of codes. The results of the calculations for ISP 22 and ISP 27 have been used to check some of these methods and proved very useful for this purpose [16, 21].

Another lesson from these small break LOCA ISPs concerns the experience gained by the code users in performing calculations on various facilities and transients, improving their understanding of the code capabilities and weaknesses. Opening this activity to Eastern countries (since ISP 22) was thus a unique opportunity specifically for small countries to have access to relevant experimental data, and to improve their know-how in relation to the use of codes and nuclear reactor safety.

A further lesson from small break LOCA ISPs concerns the identification and characterization of user effects [11]. Different code users utilizing the same code version and getting the same available information from experimentalists (ISP host organisation) produce quite different results especially in "blind" standard problems, but as well as in "open" standard problems. ISP 25 (not included in the present study) and ISP 26 (here considered) were used as basis for the influence of the user on the results of calculations (see [11]). Among the various out comings, it was found that, potentially, user effects can be very important and may overshadow code deficiencies or capabilities (same conclusion as in Section 8.1).

9. CONCLUSIONS

The ISPs are part of an important ongoing programme promoted by OECD/CSNI during the last thirty years and gave, among the other things, the possibility to disseminate the safety culture and to homogenize the knowledge of scientists from different countries of the world, in a relevant area of the nuclear technology. In addition, the ISP activity gives a real challenge to all participants to analyze an experiment in detail in the frame of an international activity and compare the own calculation results with other results (and the data). Furthermore it is a big challenge to all codes, which are used for comparing with the other codes.

The present work focuses on a limited part of the entire programme, making reference to five ISPs that deal with phenomenon typical of small break LOCAs in PWRs. Four different facilities based on experiments and an actual plant transient are involved. The considered set of standard problems represent an answer in the system thermal-hydraulic area to the concerns raised by the TMI-2 accident and have been proposed in a period when advanced codes have been made available; definitely, the discussed ISPs and the advanced codes might be considered as complementary elements for ensuring reliability in safety evaluations in the area of long lasting transients (as opposed to short transients like large break LOCA) potentially affected by operator actions.

In the frame of the presented activity, the involved experimental facilities and the reference tests have been characterized adopting the list of twenty two phenomena proposed when setting up the CSNI code validation matrix for integral test facilities. This led to establishing qualitative similarities among the different transient scenarios and demonstrated that the latest small break LOCA ISPs, which were performed in the largest scale facilities, cover much broader ranges of phenomena relevant to nuclear reactor thermal-hydraulics.

Whatever is the kind of ISP, "blind," "open," "double blind," the quality of a calculation, that is, the degree of agreement between code results and experimental data, depends upon several factors ranging from capabilities of code physical models, to user experience, to nodalization details and qualification, to the quality of the information supplied by the experimentalists, integration of this information into the input of the codes. So, as already mentioned, finalized conclusions regarding the submitted calculations cannot be drawn without the direct contributions of the code users and the experimentalists; on the other hand, this is the subject of the comparison reports issued by OECD as a summary of each ISP, they are listed here as references.

Considering the above, the conclusions reached are of a quite general nature and involve aspects that are common to the different ISPs, as well as to small break LOCA related ISPs.

It was noted that large numbers of countries (more than 20) and organisations (more than 50) took part at least in one small break LOCA ISP: these essentially include all countries using nuclear power to generate electricity (one exception strictly connected with political reasons can be observed). However, only few organisations participated in all the considered ISPs and many organisations took part in one ISP only. Furthermore, in the recent years the number of code users increased and among these users, there were less experienced ones; this must be considered carefully when deriving conclusions from the ISP activities. Assuming that the advanced codes were available to most of the participants since the time of the ISP 18 (first of the considered ISP), this together with the statistical evaluations done in the frame of Section 7 and [4], lead to the following conclusions.

(a) The objectives in the participation to the ISP changed over the time, being mostly connected with code development at the beginning and mostly focused toward user training in the latest ISP; this might not be true for codes that did not reach an adequate maturity at the beginning of the considered time frame.
(b) Notwithstanding the large effort necessary to organize or even to participate in an ISP, the cumulative experience gained by a single organisation or by a single group of scientists inside one organisation is generally not transferable or at least has not been transferred. This is especially true in a nonnegligible number of cases where the participant organisation or the group of scientists dissolved and did not leave any track of the acquired experience. This concerns code developers, experimentalists, and code users, and may be considered as a problem common to the whole area of system thermal-hydraulics.
(c) The ISPs got more demanding with the time. There was a significant progress in the code capabilities; for example, the ISP 27 (BETHSY) could be calculated only with very large difficulties (or in some cases could not be calculated at all) at the time period when the ISP 18 (LOBI) was performed.

A list of thirteen deficiencies coming from the considered ISPs and common to most of the utilized codes has been identified as in Section 8.1. This is not an exhaustive list, but underlines one positive result of ISP exercises. However, it must be observed that very slow or almost no progress has been done in the identified areas in the past decade.

An additional aspect that should be brought to the attention is that the ISPs are not part of a general finalized code assessment programme that, historically, has been the objective of cooperations like International Code Assessment Program of USNRC (ICAP), Code Assessment and Maintenance Program of USNRC, follow up to ICAP (CAMP), Club des Utilisateurs du CATHARE (CUC), and so forth or of nationally funded researches. In most of the cases, this prevented a direct improvement of codes based on the results of ISPs (see also below), although code deficiencies detected in the frame of ISPs, owing to the relevance of the ISPs themselves, were always brought to the attention of code developers.

Furthermore, inadequacy or lack of direct feedback from the results of ISPs to code model improvements is in some cases the consequence of the need to fix time frames and deadlines; this may prevent the achievement of "optimized" results with an assigned code version. For some particular codes, too frequent releases of different code versions also put obstacles as far as that feedback is concerned. The use of ISPs as exercise for proving or achieving some user qualification, also contributed to the above conclusion.

Although a detailed evaluation/judgment of each ISP activity is not the purpose of the effort done in the present framework, it seemed worthwhile to add few specific conclusions applicable to single ISPs.

(i) A large mismatch may exist between the huge effort from the host organisation and the participants as a whole on one side, and the final result of the exercise.

(ii) Incomplete or even misleading information supplied by the host organisation in some cases testify of the complexities of the general code assessment problem and could hinder to facilitate the achievement of meaningful conclusions.

(iii) In some cases, participants underestimated the effort necessary to set up suitable nodalization including correct consideration of initial and boundary conditions; this constitutes an additional reason preventing more satisfactory conclusions of the activities.

(iv) Especially, as a consequence of the above, quite vague formulations can be found in the general conclusions of the ISP reports.

(v) A large range of results obtained by participants using the same code version gives interesting information about uncertainty in selection of input parameters and uncertainties of code models as well as experimental data errors (see [11]).

9.1. Recommendations

General recommendations coming from the performed activity can be summarized as follows, covering different aspects connected with small break LOCA ISPs.

(i) The participation into ISP activities of non-OECD countries should be continuously encouraged; especially small countries not having the capabilities for wide national research programmes, can get substantial benefits from ISPs.

(ii) Notwithstanding obvious drawbacks (e.g., lack of suitable instrumentation, inaccuracy of data base, etc.) a future ISP based on an actual plant transient, if any, is highlyrecommended.

(iii) A better characterization of the experiments of ISPs, also in view of a qualitative evaluation of code performance, could be based on the 67 phenomena identified for the CSNI separate effects tests code validation matrix made available in mid 90 s [22, 23], future ISPs should directly consider this.

(iv) The interaction between ISP host/proposing organisation and CSNI working groups has been quite satisfactory as far as the test selection is concerned, but could be improved especially in relation to the evaluation of the results and for defining the impact of these in the thermal-hydraulic and nuclear safety areas.

(v) The inadequacy of a direct feedback (indirect feedback may exist) between ISPs results and code developers has already been stressed. However, indirect feedback exists, as ISPs revealed the important role played by physical phenomena such as phase separation at the junctions, stratification in horizontal components (ISP 18), or secondary side heat transfer (ISP 27). Then, valuable information for improving the code model must be the result of independent confirmatory analyses performed utilizing data from separate effects tests facilities (SETF), for example, a code inadequacy possibly identified when performing the analysis of one ISP in an integral test facility should be confirmed and characterized by calculations based on SETF experiments. In this sense, SETF-based ISPs are also strongly recommended.

(vi) The list of code deficiencies given in the Section 8.1 could be used as basis for planning future ISPs in separate effects tests facilities together with phenomena relevant in 2D/3D geometrical configurations. Clearly, codes should also be improved as far as possible, when a model inadequacy is found.

(vii) "Blind" types of ISPs should be preferred to "Open" types, especially when a posttest ("Open") phase of the ISP can be planned and reliable data can be supplied to the participants since the beginning. This gives a better opportunity to evaluate the user effect and better represents the overall situation that is faced when performing plant related calculations.

(viii) The experience acquired so far, the database available from different national and international programmes and the cost of an ISP, suggests not to propose additional ISPs in the frame of small break LOCAs; transients evolving at low pressure, scenarios involving complex accident management procedures or of specific interest for the new generation reactors are not part of this recommendation.

(ix) Some of the discussed ISPs have been utilized as sample basis for addressing the problems of user effects and quantification of the accuracy of calculation results. However, some specific efforts should be devoted from future ISP host organisations, possibly in cooperation with CSNI, in the areas of user effects, user qualification, and quantification of the accuracy. It could even be standard part of the ISP activity.

(x) In relation to user effect, in a long-term view, a part of the problem can be solved by improved codes, which remove the need for the user to make ad hoc assumptions in order to compensate for code limitations or complete lack of modelling; an example of this is modelling pressure drop at geometric discontinuities.

(xi) In connection with the above, when applicable, the problem of evaluating the uncertainty by system thermal-hydraulic codes when predicting scenarios relevant to nuclear power plants could be addressed in the frame of activities similar to the ISPs.

Finally, considering the effort expended in the preparation of ISPs, it would be very useful if this information was catalogued and stored so that it could be easily accessed for future posttest analyses.

NOMENCLATURE

A_b:	Broken area size of steam generator tubes
A_{max}:	Maximum area size of steam generator tubes
ACC:	Accumulators
BAF:	Bottom of active fuel
BL:	Broken loop
CAMP:	Code Assessment and Application Programme of U.S. NRC
CEA:	Commissariat pour l'Energie Atomic
CEC:	Commission of European Community
CENG:	Centre d'Etudes Nucleaires Grenoble (present name: CEA Grenoble)
CL:	Cold leg
CSNI:	Committee on the Safety of Nuclear Installations
CUC:	Cub des Utilisateur du CATHARE
D:	Diameter
ECC:	Emergency core cooling
EFW:	Emergency feed water
ENEA:	Ente nazionale energie alternative
HPIS:	High-pressure injection system
ICAP:	International Code Assessment Program of U.S. NRC (predecessor of CAMP)
IL:	Intact loop
ISP:	International standard problem
JAERI:	Japan Atomic Energy Research Institute
JRC:	Joint European Centre
K_v:	Volume scaling factor
L:	Length
LOCA:	Loss-of-coolant accident
LOFW:	Loss of feed water
LPIS:	Low-pressure injection system
MSIV:	Main steam isolation valve
NEA:	Nuclear energy agency
OECD:	Organisation for Economical Cooperation and Development
PORV:	Power operated relief valve
PRZ:	Pressurizer
PS:	Primary side
PSI:	Paul Scherrer Institut
PWG-2:	Principal working group on system behaviour
PWR:	Pressurized water reactor
RHR:	Residual heat removal
SBLOCA:	Small break LOCA
SG:	Steam generator
SGTR:	Steam generator tube rupture
SI:	Safety injection
SRV:	Safety relief valve
SS:	Secondary side
TAF:	Top of active fuel
TG-THSB:	Task Group on Thermal-Hydraulic System Behaviour
TMI-2:	Three Mile Island Unit 2

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